Written Exam Question Worksheet

Examination Outline Cross-Reference Tier # 2

Evolution/System 003 Reactor Coolant Pump System (RCPS) **Group #** 1

K/A # K3.04 **Page #** 3.4-6 **RO/SRO Importance Rating** 3.9 4.2

Measurement Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: RPS.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 55.43

Proposed Question ✔ RO SRO ✔ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor operating at 50% power, with ICS in automatic.
- RPS Channel 1C is de-energized due to loss of Vital Bus 1C (Inverter 1C failure).
- All four Reactor Coolant Pumps operating.
- Core power imbalance is negative 3%.

Event:
- Reactor Coolant Pump RC-P-1D trips due to breaker fault.

Based on these conditions, identify the ONE selection below that describes:
(1) Response of the Reactor Protection System to this event.
(2) Reason for the RPS response.

A. (1) Channel D (only) trips.
(2) Nuclear overpower based on RCP operating status.

B. (1) Channel D (only) trips.
(2) Nuclear overpower based on RCS Flow and Imbalance.

C. (1) Channels A, B and D trip.
(2) Nuclear overpower based on RCP operating status.

D. (1) Channels A, B and D trip.
(2) Nuclear overpower based on RCS Flow and Imbalance.

Technical Reference Lesson Plan 11.2.01.132, Reactor Protection System , Page 21, Rev. 17.
Technical Specifications Table 2.3-1, RPS Trip Setting Limits, Note (4), Page 2-10, Amendment 247.

Open Exam Reference None.

Learning Objective IV.E.14.37

Question Source New Bank **Question #**

✔ Modified Bank **Parent Question #** QR4E14-25-Q02

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because all 3 remaining channels will trip.

Distracter is plausible because RPS Channel D will, in fact, trip due to high flux-to-pumps. This distracter presents the correct response of RPS Channel D (only).

B INCORRECT because all 3 remaining channels will trip due to (false) trip of BOTH RCS Loop B RCPs, and trip will be due to actuation of the Flux-to-Pumps circuit. Core imbalance is not negative enough to reduce the high flux trip setpoint.

Distracter is plausible because RCS flow will be reduced due to trip of RC-P-1D.

C CORRECT. All 3 remaining RPS channels trip due to false actuation of the high flux trip based on RCP operating status. The Pump Power monitor trip erroneously detects that both RCS Loop B RCPs are not operating (de-energizing VBC results in loss of RC-P-1C power monitor - false output state indicates no motor power, bistable de-energizes signals to all 4 RPS channels).

D INCORRECT because the channels will trip due to (false) trip of BOTH RCS Loop B RCPs, and trip will be due to actuation of the Flux-to-Pumps circuit. Core imbalance is not negative enough to reduce the high flux trip setpoint.

Distracter is plausible because RCS flow will be reduced due to trip of RC-P-1D.

Comments None.

contacts associated with that channel open, and the RPS contact monitor receives a loss of pump signal.

f. The RPS contact monitor provides an analog signal based on pump status as the input for the power-to-pumps trip bistable which initiates a protection channel trip for the conditions.

g. Reactor Coolant Pump Power Monitors

1) Two redundant monitors (trains) per RCP motor (Rack A-red), Rack B-green).

2) Either pump monitor channel trip will actuate all four RPS channel contacts.

3) Each RCP power monitor channel (in both racks) is powered from respective vital bus; i.e., RC-P-1A power monitor (in both racks) is powered from VBA.

4) Loss of vital bus will actuate that RC pump's power monitor, and all four RPS channels will see that RC pump as tripped; i.e., loss of VBA will tell all four RPS channels that RC-P-1A is tripped.

5) Operation of pump power monitor.

a) Watts transducer senses power drawn by RCP motor (normally ~6MW)

b) Output of transducer or test input (SS7) goes through a wattmeter to a bistable which will trip at <25% load (2.1 MW).

c) Trip bistable will actuate a 0.55 second timer.

d) Trip of timer will open circuit from vital bus to de-energize output relays for pump (62X), as long as SS1 keyswitch is in "normal" position.

e) The SS1 (Bypass/Normal) switch is a "break before make" contact switch. Therefore as the switch is manipulated, the output logic

TABLE 2.

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS (5)

	Four Reactor Coolant Pumps Operating (Nominal Operating) Power – 100%	Three Reactor Coolant Pumps Operating (Nominal Operating) Power - 75%	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, max. % of rated power	105.1	105.1	105.1	5.0(2)
2. Nuclear power based on flow (1) and imbalance max. of rated power	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Power/Flow Setpoint in COLR times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (4) on pump monitors max. % of rated power	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig max.	2355	2355	2355	1720(3)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Reactor coolant temp. F., max.	618.8	618.8	618.8	618.8
7. High Reactor Building pressure, psig max.	4	4	4	4
8. Variable low reactor coolant system pressure, psig min.	(16.25 T_{out} - 8113)(6)	(16.25 T_{out} - 8113)(6)	(16.25 T_{out} - 8113)(6)	Bypassed

(1) Reactor coolant system flow, %
(2) Administratively controlled reduction set during reactor shutdown.
(3) Automatically set when other segments of the RPS (as specified) are bypassed.
(4) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation.
(5) Trip settings limits are limits on the setpoint side of the protection system bistable connectors.
(6) T_{out} is in degrees Fahrenheit (F).

Form ES-401-5 **Written Exam Question Worksheet** Q # 002

Examination Outline Cross-Reference Tier # 2

Evolution/System 003 Reactor Coolant Pump System (RCPS) Group # 1

K/A # K4.04 Page # 3.4-7 RO/SRO Importance Rating 2.8 3.1

Measurement Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motor and seals.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 55.43

Proposed Question ✓ RO SRO ✓ PRA Related **Correct Answer** C.

In accordance with OP-TM-226-000, Reactor Coolant Pumps, identify the ONE condition below that requires the operator to trip all 4 RCPs.

A. Operating Makeup Pump trips.

B. Major steam leak environment in the RB.

C. NS-V-15, RB NS Cooling Isolation Valve (MOV), closes due to a circuit failure.

D. Operating Intermediate Closed Cooling Pump trips, and the standby pump fails to start.

Technical Reference OP-TM-226-000, 2.2.8 Dash #2, Page 4, Rev. 1.

Open Exam Reference None.

Learning Objective IV.D.05.24

Question Source ✓ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✓ Memory/Fundamental Knowledge Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because intermediate closed cooling provides cooling to thermal barrier heat exchangers ensuring RCPs seals are cooled.

Plausible because loss of the operating makeup pump will result in a loss of seal injection flow to the RCPs.

B INCORRECT because a steam leak in the RB will not require a trip of the RCPs.

Plausible because a major steam leak could potentially result in over heating of the pump motors and bearings.

C CORRECT. Nuclear Service Closed Cooling is lost to all 4 RCPs.

D INCORRECT because while Intermediate Closed Cooling is lost to the RCP thermal barriers, Seal injection is still operating to force cool water through the seals.

Plausible because loss of IC (in conjunction with low RCP seal injection flow) is included in an interlock to automatically trip or require manual trip of all 4 RCPs.

Comments None.

2.2.8 To avoid or limit component damage, shutdown the affected RC Pump for any of the following:

- Total Loss of Seal Injection and Intermediate Cooling.
- Loss of NS cooling flow (All RCPs for a total loss of NS to RB)
- Motor bearing Upper or Lower Guide temperatures exceed 185°F.
- Motor Thrust bearing (Up or Down) temperatures exceed 195°F.
- Motor winding temperature exceeds 302°F (150°C).
- Pump bearing temperature exceeds 225°F.
- Number 1 Seal Inlet temperature exceeds 225°F.
- Number 1 Seal Leak-Off flow is > 6 gpm at normal operating pressure
- Number 1 Seal Leak-Off flow is < 0.8 gpm at normal operating pressure
- Number 2 Seal Leak-Off flow is excessive at normal operating pressure, as evidenced by RCDT level rise is > 1 gpm attributable to a RCP **and** one of the following conditions exist on that RCP:
  - High Standpipe Level alarm
  - RCP Pump or Motor vibration rising
- Pump Vibration: exceeds 20 mils with 4-pump operation, or 30 mils with single pump operation.
- Motor vibration exceeds 7 mils.

2.2.9 To avoid excessive leakage, do **not** allow RCS Pressures > 40 psig with any RC Pump uncoupled from the motor.

2.2.10 To avoid motor damage during uncoupled runs, do **not** operate motor with vibration measured at the pump vibra-switch exceeding 2 mils.

2.2.11 To avoid seal damage, Seal injection water flow is required to all reactor coolant pumps when reactor coolant temperature is above 190°F **and** pressure is above 100 psig, except when operating in the loss of injection mode.

2.2.12 To avoid damage to Number 2 seal, maintain Number 2 seal inlet pressure between 27 to 90 psig as read on MU-39-PI-1.
Target pressure for optimal seal performance is 40 to 60 psig (MU-39-PI-1).

Form ES-401-5 **Written Exam Question Worksheet** Q # 003

Examination Outline Cross-Reference Tier # 2

Evolution/System 004 Chemical and Volume Control System Group # 1

K/A # K6.14 Page # 3.2-11 RO/SRO Importance Rating 3.1 3.3

Measurement Knowledge of the effect of a loss or malfunction on the following CVCS components: Recirculation path for charging pumps.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 55.43

Proposed Question ✔ RO SRO PRA Related **Correct Answer** A.

Initial plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- Makeup Pump Recirc Valve MU-V-36 is tagged (valve open, breaker open) out of service for maintenance.

Event:
- Reactor trip due to low RCS pressure.
- Loss of off-site power (LOOP)
- Emergency Diesel Generator EG-Y-1B FAILED to start.
- Channel A and B ES Actuation.
- Makeup System Flow Data:
  - MU-V-16A HPI flow is 200 gpm.
  - MU-V-16B HPI flow is 200 gpm.
  - RCP seal injection flow is 35 gpm.
  - Makeup Pump MU-P-1A recirculation flow (last measured) is 100 gpm.

Identify the ONE selection below that completes the following statement describing the effect of this malfunction on operation of the HPI system:

For purposes of throttling HPI, in accordance with OP-TM-EOP-010, current HPI flow equals

_______________.

A. 535 gpm.

B. 500 gpm.

C. 435 gpm.

D. 400 gpm.

Technical Reference OP-TM-EOP-010 Rule 2, SCM, Step A.1, Page 5, Rev. 3.
11.2.01.069, MU System, Section II.B.7.m, Page 27, Rev. 27.

Open Exam Reference None.

Learning Objective IV.A.09.51

Question Source ✔ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer. For this case, HPI flow is the sum of all pump flows (MU-V-16A/B HPI, RCP Seal Injection, and pump recirculation flows). The goal of HPI throttling is to maximize flow, and still limit Makeup PUMP flow to prevent runout. Since recirculation line flow has not been isolated in this question, this flow needs to be added when determining total HPI Pump flow for purposes of throttling HPI in accordance with OP-TM-EOP-010. In addition, RCP Seal Injection flow is required to be included in this calculation.

B INCORRECT answer because this value does not include RCP Seal Injection flow. For this case, HPI flow is the sum of all pump flows (MU-V-16A/B HPI, RCP Seal Injection, and pump recirculation flows).

Distracter is plausible because RCP Seal injection flow is injected into the RCP Seal packages, rather than through the HPI nozzles.

C INCORRECT because this answer does not include the 100 gpm pump recirculation flow. For this case, HPI flow is the sum of all pump flows (MU-V-16A/B HPI, RCP Seal Injection, and pump recirculation flows).

Distracter is plausible because the 100 gpm recirculation flow is diverted back to the Makeup Tank, rather than through the HPI nozzles.

D INCORRECT answer because this value does not include RCP Seal Injection flow or pump recirculation flow. For this case, HPI flow is the sum of all pump flows (MU-V-16A/B HPI, RCP Seal Injection, and pump recirculation flows).

Distracter is plausible because RCP Seal injection flow is injected into the RCP Seal packages, and recirculation flow is diverted back to the Makeup Tank, rather than through the HPI nozzles.

Comments None.

HPI

2

Rule 2

HPI/LPI Throttling

A. A. **IAAT** HPI has been initiated, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** MU Pump flows ≤ 515 GPM/pump.	**THROTTLE** HPI to between 500 and 515 GPM/pump.
2. **VERIFY** SCM < 250 °F	**THROTTLE** HPI to control SCM < 250°F
3. **VERIFY** at least one RCP is operating.	**THROTTLE** HPI to control SCM between 30°F and 70°F.
4. **When** incore temperature is reducing **and** RCS < 25°F superheat **and** either of the following conditions exists ____ SCM > 25°F ____ LPI > 1250 GPM in each line **then** HPI **may** be THROTTLED to < 500 GPM/pump.	

B. **IAAT** HPI is THROTTLED **and <u>not</u>** in "Piggy Back", **then OPEN** MU-V-36 **and** MU-V-37 as required to maintain MU pump flow > 115 GPM.

C. **IAAT** LPI has been initiated, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** DH Pump flows ≤ 3300 GPM (as indicated on DH-FI-802A/803A).	**THROTTLE** LPI to the maximum controllable flow ≤ 3300 GPM.
2. **If** BS-P-1 on same ES train is operating **and** taking suction from the RB sump, **then VERIFY** DH Pump flows < 3000 GPM (as indicated on DH-FI-802A/803A).	**THROTTLE** LPI to the maximum controllable flow ≤ 3000 GPM.

(I.e., ALTERNATE pump is STARTED BEFORE operating pump is SHUTDOWN.)

m. Makeup Pump Recirculation Flow Rate

A recirculation line designed for 95 gpm flow rate is required for each makeup pump.

Basis: A minimum continuous makeup pump flow rate of 135-140 gpm was desired to obtain the longest pump lifetime. The mechanical seal design, the seal injection flow requirement is 32 gpm. The normal system flow demand on the makeup pump then became ~50 gpm (45 gpm normal letdown plus ~4 gpm RC pump controlled bleedoff). Thus, the recirculation line orifice was sized for 95 gpm to increase the normal makeup pump flow rate to 145 gpm.

Note: The 95 gpm recirculation line flow rate is not the minimum allowable pump flow rate. The minimum allowable pump flow rate is 40 gpm. Operating time at 40 gpm should be minimized and should only occur when the recirculation flow path is not available. The 40 gpm removes sufficient heat from the pump to prevent the pumped fluid from vaporizing. Fluid vaporization could cause the pump internals to bind or seize.

n. Makeup Line Flow Rate Capacity

1) The makeup line flow rate capacity (including makeup control valve MU-V17) in conjunction with seal injection inleakage to the RCS is required to provide a minimum total flow rate of 140 gpm to the RCS at normal RCS operating pressure.

2) The maximum makeup line flow rate with MU-V-217 full open , two MU pumps operating and RCS pressure at 2155 psig is 333 GPM. This requirement is met by throttling MU-V-222 to 1 ¼ turns OPEN, This valve position was demonstrated by test. This requirement is an assumption for the MU pump NPSH analysis.

3) Basis: The min flow requirement is to maintain a constant reactor coolant inventory (constant pressurizer water level) and normal seal injection

Form ES-401-5 **Written Exam Question Worksheet** Q # 004

Examination Outline Cross-Reference Tier # 2

Evolution/System 004 Chemical and Volume Control System Group # 1

K/A # K2.05 Page # 3.2-8 RO/SRO Importance Rating 2.7 2.9

Measurement Knowledge of bus power supplies to the following: MOVs.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 55.43

Proposed Question ☑ RO SRO PRA Related **Correct Answer** B.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- 1C 480V ES Valves Motor Control Center is powered from its normal power supply.

Event:
- Reactor trip due to high RCS pressure.
- Failed open Pressurizer safety valve resulted in ES actuation.
- 1P 480V Bus trip due to electrical fault at the time of the ES actuation.

Based on these conditions identify the ONE selection below that completes the following statement:
HPI pump suction from the BWST will be supplied through __________.

A. BWST suction valve MU-V-14A (only).

B. BWST suction valve MU-V-14B (only).

C. Both BWST suction valves MU-V-14A and MU-V-14B.

D. Both Decay Heat Pump Piggy-back Valves DH-V-7A and DH-V-7B.

Technical Reference 1107-5 , Electrical Distribution Panel Listing, Makeup and Purification Section 46, Page 75, Rev. 123.

Open Exam Reference None.

Learning Objective IV.A.09.27

Question Source ✔ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. MU-V-14A is closed and will not open under these conditions. MU-V-14A power supply is 1A 480V ES MCC, which is powered from 1P 480V Bus. MU-V-14A is closed during normal power operations.

Distracter is plausible based on the possibility of reversing the two valve power supplies.

B CORRECT. MU-V-14B would open since its power supply is 1B 480V ES Valves MCC, which is powered from 1S 480V Bus. MU-V-14A would remain closed due to loss of 1P 480V Bus power.

C INCORRECT. MU-V-14A would remain closed due to loss of 1P 480V Bus power.

Distracter is plausible based on the misconception that both of these valves are powered from the swing bus 1C ES Valves MCC mentioned in the stem. This misconception is plausible because the suction cross-connect valves are all open during normal power operations, making the suction header common to all 3 HPI pumps.

D INCORRECT. MU-V-14B would open since its power supply is 1B 480V ES Valves MCC, which is powered from 1S 480V Bus. Also DH-V-7A would not have power due to loss of 1P 480V Bus.

Distracter is plausible because DH-V-7A and DH-V-7B provide possible flowpaths from the BWST using the DHP pumps. These two valves are used for "piggy-back" operation - DHPs supply HPI Pump suction from RB Sump when BWST is depleted.

Comments None.

46. MU (MAKE-UP AND PURIFICATION)

TAG NO.	VALVE OP	DESCRIPTION	CONTROL DRAWING	ELECTRICAL DISTRIBUTION	UNIT/ SW
MU-K-1A		MU AND DEMINERALIZER A	NON-ELECT		
MU-K-1B		MU AND DEMINERALIZER B	NON-ELECT		
MU-P-1A		MAKEUP PUMP A	208-213,	1D 416OV	1D7
MU-P-1B		MAKEUP PUMP B	208-215,	1D 416OV	1D8
			216	1E 416OV	1E9
MU-P-1C		MAKEUP PUMP C	208-214	1E 416OV	1E8
MU-P-2A		MU-P-1A AUX OIL PUMP	208-523	1A-ESV CC	6C
MU-P-2B		MU-P-1B AUX OIL PUMP	208-523	1A-ESV CC	4A
MU-P-2C		MU-P-1C AUX OIL PUMP	208-523	1B-ESV CC	6C
MU-P-3A		MU-P-1A MAIN OIL PUMP	208-562	1A-ES CC	14C
MU-P-3B		MU-P-1B MAIN OIL PUMP	208-563	1B-ESV CC	6A
MU-P-3C		MU-P-1C MAIN OIL PUMP	208-562	1B-ES CC	2A
MU-P-4A		MU-P-1A GEAR OIL PUMP	208-647	1A-ESV CC	10B
MU-P-4B		MU-P-1B GEAR OIL PUMP	208-648	1C-ESV CC	1E
MU-P-4C		MU-P-1C GEAR OIL PUMP	208-649	1B-ESV CC	1D
MU-V-1A	MO	LET DOWN COOLER INLET	208-435	1A RAD WCC	6A
MU-V-1B	MO	LET DOWN COOLER INLET	208-435	1B-RW DCC	6A
MU-V-2A	MO	LET DOWN COOLER OUTLET	208-437	1B-ESV CC	4D
MU-V-2B	MO	LET DOWN COOLER OUTLET	208-498	1B-ESV CC	5D
MU-V-3	P	LET DOWN ISOL. AT CONT. VESSEL	209-022	RSTSP-A	*FU-MU1(AQ)
			(SH. 1)		*FU-MU2(AR)
MU-V-4	P	LET DOWN ISOL. AT LET DOWN ORIFICE	209-020	XCC	FA 7,8
MU-V-5	D	LET DOWN FLOW CONT. BYPASS	302-660	NON-ELEC	
MU-V-6A	P	LET DOWN FLOW ISOL. INLET TO MU DEMIN.	209-020	XCC	FA 3,4
MU-V-6B	P	LET DOWN FLOW ISOL. INLET TO MU DEMIN.	209-020	XCC	FA 1,2
MU-V-8	MO	LET DOWN SPLIT TO FILTERS			
		OR HOLD UP TANKS	208-438	1A-RW MCC	6B
MU-V-9	D	MAKE UP ADD. TO LETDOWN STREAM	302-66	NON-ELEC	
MU-V-10	P	MAKE UP ADD. TO LETDOWN ISOL.	209-021	XCC	FD 71,72
MU-V-11A	P	INLET ISOL. VALVE TO MAKE UP FILTERS	209-021	XCC	FA 9,10
MU-V-11B	P	INLET ISOL. VALVES TO MAKE UP FILTERS	209-020	XCC	FA 5,6
MU-12	MO	MAKE UP TANK DISCHARGE ISOL.	208-439	1B-RW MCC	6C
MU-V-13	D	MAKE UP TANK VENT	209-080	XCC	FA 14,15
MU-V-14A	MO	MAKE UP PUMP SUCTION FROM BWST	208-440	1A-ES	7D
MU-V-14B	MO	MAKE UP PUMP SUCTION FROM BWST	208-440	1B-ESV CC	4A
MU-V-16A	MO	H.P. INJECT. ISOL. AT CONT. VESSEL	208-442	1A-ESV CC	4B
MU-V-16B	MO	H.P. INJECT. ISOL. AT CONT. VESSEL	208-442	1A-ESV CC	4C
MU-V-16C	MO	H.P. INJECT. ISOL. AT CONT. VESSEL	208-442	1B-ESV CC	4B
MU-V-16D	MO	H.P. INJECT. ISOL. AT CONT. VESSEL	208-442	1B-ESV CC	4C
MU-V-17	D	CHARGING LINE FLOW CONT. VALVE	302-661	NON-ELEC	
MU-V-18	P	CHARGING ISOL. AT CONT. VESSEL	209-022	RSTSP-B	FU-MU1(CH)
			(SH. 2)		FU-MU2(CH)
					FU-MU5(CM)
					FU-MU6(CM)
		*2 Fuses Each			

Form ES-401-5 **Written Exam Question Worksheet** Q # 005

Examination Outline Cross-Reference Tier # 2

Evolution/System 005 Residual Heat Removal System Group # 1

K/A # K6.03 Page # 3.4-12 RO/SRO Importance Rating 2.5 2.6

Measurement Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 55.43

Proposed Question ✓ RO SRO ✓ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor is in Cold Shutdown condition.
- Decay Heat Removal Train A is operating.
- Decay Heat Closed Cooling flow through the Decay Heat Removal cooler is throttled to maintain the RCS at 130 degrees F.
- Total loss of Instrument Air (0 psig) occurred.

Identify the ONE statement below that describes the response of the cooling system and subsequent effect on RCS temperature for this situation.

A. Closure of DC-V-65A (Cooler bypass) AND DC-V-2A (Cooler inlet) results in RCS heatup.

B. Opening of DC-V-65A (Cooler bypass) AND DC-V-2A (Cooler inlet) results in RCS cooldown.

C. Closure of (ONLY) DC-V-2A (Cooler inlet) results in RCS heatup.

D. Opening of (ONLY) DC-V-2A (Cooler inlet) results in RCS cooldown.

Technical Reference 1202-36, Loss of Instrument Air, Page 6, Rev. 34.

Open Exam Reference None.

Learning Objective IV.D.14.11

Question Source New ✓ Bank Question # Bank - May 2003 TMI SRO Exam - Q-051.

Modified Bank Parent Question #

Question NRC Exam History TMI 2003 Q-051

Question Cognitive Level Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because DC-V-2A fails open on loss of IA.

Plausible distracter since DC-V-65A fails closed on loss of IA, and closure of both valves (incorrect response for loss of Instrument Air) will result in loss of cooling and therefore RCS heatup.

B INCORRECT answer because DC-V-65A fails closed on loss of IA.

Plausible distracter since DC-V-2A fails open on loss of IA, and opening both valves (incorrect response for loss of Instrument Air) would result in RCS cooldown.

C INCORRECT answer because DC-V-2A fails open on loss of IA.

Plausible distracter since closure of DC-V-2A (incorrect response for loss of Instrument Air) would result in RCS heatup.

D CORRECT answer.

Comments May 2003 TMI SRO Exam - Question 051.

Effect on RHRS is addressed in Answer D - Cooling water inlet valve will fail wide open. RCS cooldown results from the failure. Added (ONLY) to answers C and D.

_____ Open MU-V-110

_____ 13.2 Dispatch an auxiliary operator to MU-V-3 and establish communication with the Control Room.

_____ 13.3 Monitor letdown flow and prefilter ΔP and slowly re-establish letdown flow at 2.5 gpm/min. (unless waived by the SM/CRS) by opening MU-V-3.

14. If a normal seal return flowpath is desired, perform the following:

_____ 14.1 Dispatch an auxiliary operator to MU-V-26 and establish communication with the Control Room.

_____ 14.2 Locally open MU-V-26.

_____ 15. Upon verified loss of both Seal Injection and Intermediate Cooling Water to the RC pump seals:

_____ A. Verify tripped or trip the Reactor.

_____ B. Verify tripped or trip the Reactor Coolant Pumps.

_____ C. Go to OP-TM-EOP-001 and refer to 1202-36 for Loss of Instrument Air.

_____ 16. If Reactor is shutdown, verify $\geq$ 1% ΔK/K shutdown margin. Use BWST and MU-V-14A to borate as necessary.

_____ 17. If on DHR, Monitor DHR suction and return temperatures throttle flows or secure pumps as necessary to minimize RCS temperature transient. DC-V-2's fail open and DC-V-65's fail closed providing full DC flow through the cooler.

NOTE

The turbine jacking gear must be manually engaged locally when IA-V-26 is shut.

_____ 18. Verify IA-V-26 closed if pressure is below 60 psig.

_____ 19. If CO-V-8 fails open and condenser hotwell level is high, close CO-V-13.

_____ 20. If Decay Closed Surge Tanks indicate high level verify DC-V-19A and B failed open due to loss of air and close DC-V-20A and B to isolate surge tank makeup.

Form ES-401-5 **Written Exam Question Worksheet** Q # 006

Examination Outline Cross-Reference Tier # 2

Evolution/System 006 Emergency Core Cooling System (ECCS) Group # 1

K/A # K4.20 Page # 3.2-17 RO/SRO Importance Rating 3.2 3.5

Measurement Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Automatic closure of common drain line and fill valves to accumulator.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor operating at 100% power, with ICS in full automatic.
- CF-T-1A level has been raised using OP-TM-213-469, Makeup to CF-T-1A for Batches of Less Than 100 Gallons.
  - Fill isolation valve CF-V-19A has not yet been closed.
- Chemistry is initiating sampling of CF-T-1A using N1807, Primary Chemistry Sampling.
  - Sample isolation valve CF-V-20A is open to support their sample for final boron concentration verification.

Event:
- Automatic reactor trip due to low RCS pressure.
- RCS leak (LOCA) inside the RB.
- NO ES actuations have occurred yet.

Based on these conditions identify the ONE selection below that describes when to expect automatic closure of CF-V-19A and CF-V-20A.

A. CF-V-19A will close when Reactor Trip Isolation occurs;
CF-V-20A will close when Reactor Trip Isolation occurs.

B. CF-V-19A will close when Reactor Trip Isolation occurs;
CF-V-20A will close when 4 psig ES actuation occurs.

C. CF-V-19A will close when 4 psig ES actuation occurs;
CF-V-20A will close when Reactor Trip Isolation occurs.

D. CF-V-19A will close when 4 psig ES actuation occurs;
CF-V-20A will close when 4 psig ES actuation occurs.

Technical Reference 209-023, CF-V-20A, Rev. 5.
209-024, CF-V-19A and CF-V-19B, Rev. 4.
11.2.01.014, Core Flood System, Pages 5 and 6, Rev. 13.

Open Exam Reference None.

Learning Objective IV.A.13.4

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A Correct. Both CF-V-19A and CF-V-20A close on Reactor Trip Isolation signals as part of diverse means of containment isolation for accident conditions. Both valves also receive a back-up closure signal from 4# ES actuation.

B Incorrect because CF-V-20A closes on Reactor Tip Isolation.

Plausible because CF-V-19A closes on Reactor Trip Isolation, and CF-V-20A (back-up) closes on a 4# ES actuation.

C Incorrect because CF-V-19A closes on Reactor Tip Isolation.

Plausible because CF-V-20A closes on Reactor Trip Isolation, and CF-V-19A (back-up) closes on a 4# ES actuation.

D Incorrect because both CF-V-19A and CF-V-20A close on Reactor Trip Isolation.

Plausible because CF-V-19A and CF-V-20A close (back-up) on 4# ES actuation.

Comments None.

REFERENCE DWGS.:

INDEX SS-209-001 THRU 004
LEGEND SS-208-001
LOGIC DIAGRAM. S-203-817
SS DEVELOPMENT: SS-209-007

NOTES:

1. ▲▲ = SAMPLING PANEL
2. 20X/CF-V20A: CLARK 404-2

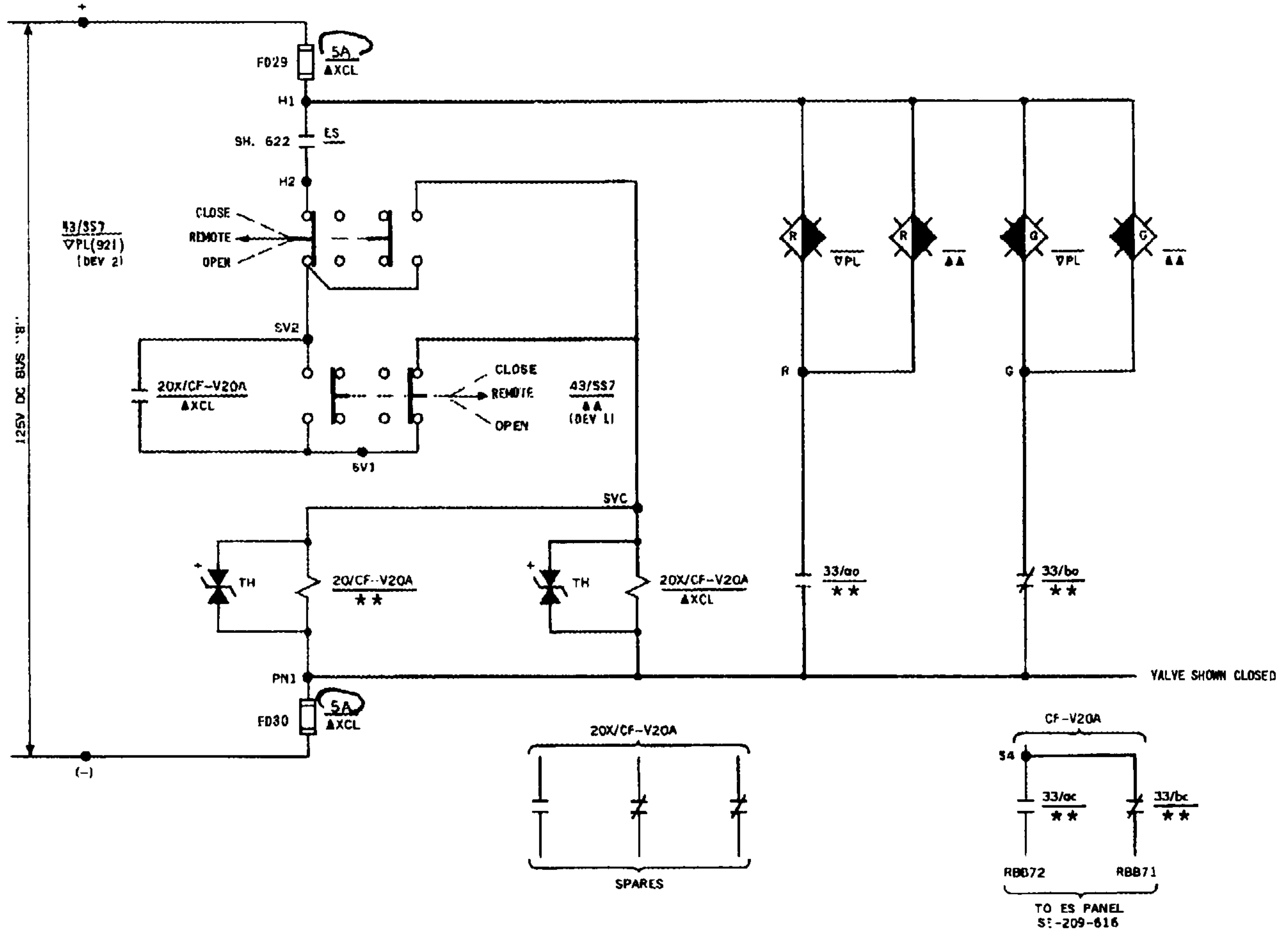
METROPOLITAN EDISON COMPANY THREE MILE ISLAND NUCLEAR STATION UNIT 1	MADE P.D. CHK'D. FFB SQ. CF. GC CF. DFN. JG ENG. W.F. Zeller	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.	
ELECTRICAL ELEMENTARY DIAGRAM D.C. & MISCELLANEOUS		4192 WORK ORDER	SS-209-023 DRAWING · 5 REV.

REV. CH. APP. DATE: 1-KG FB 4-8-71; 2-KG-FB 10-1-71; 3-DP-FB 4-12-72; 4 SEE DCR 2-1023 KG FB PJS 6-11-73; 5 SEE DCR 2-1100 KG FB PJS

B

CONSTRUCTION / BIDDING PURPOSES ONLY / RELEASED FOR / ENGR. / DATE

CORE FLOODING TANK "A" SAMPLE VV. CF-V20A



REFERENCE DWGS.:
LEGEND: SS-208-001
LOGIC DIAGRAM: S-203-B18
INDEX: SS-209-001 THRU 004
SEL. SW. DEVELOPMENT SS-209-007

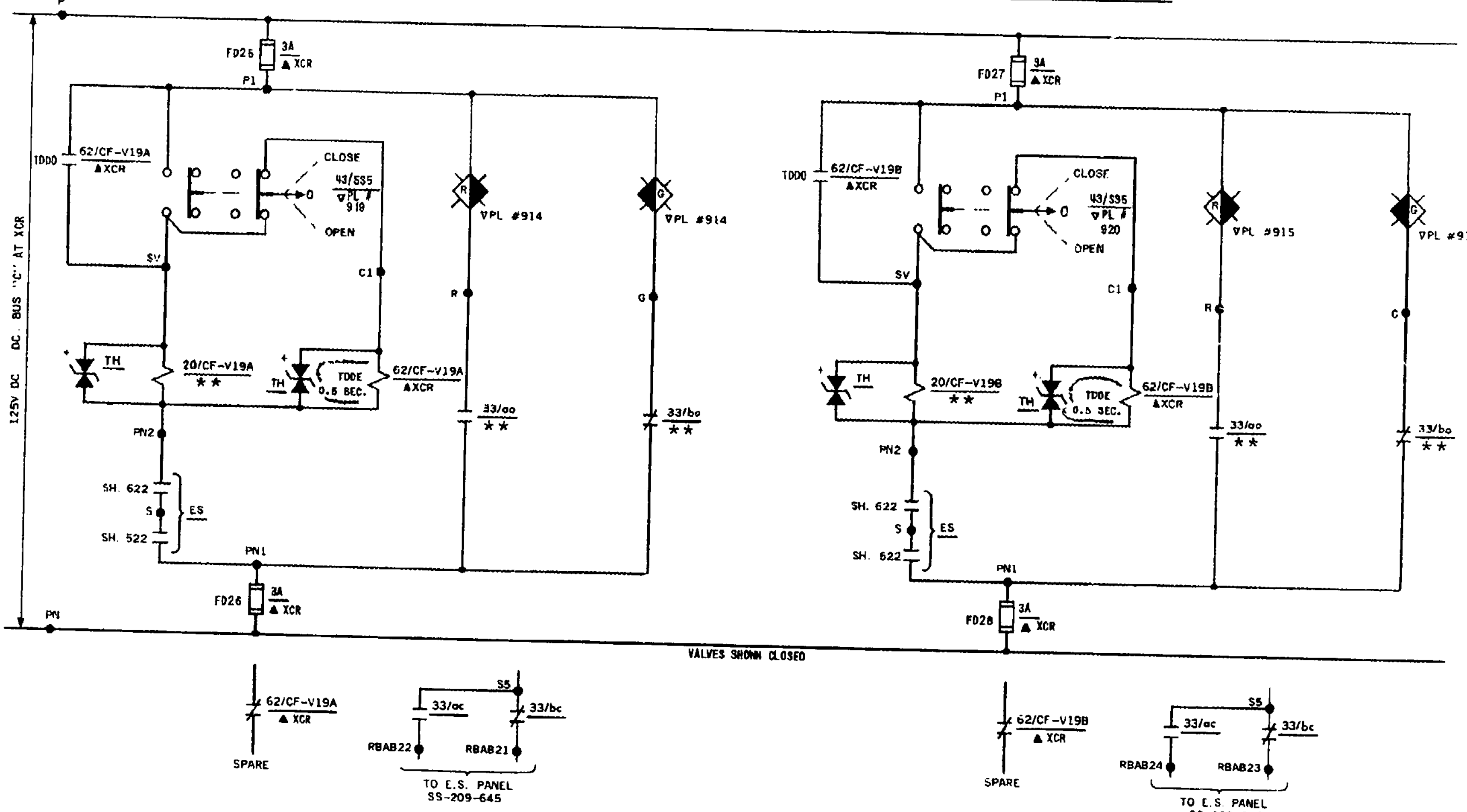
METROPOLITAN EDISON COMPANY THREE MILE ISLAND NUCLEAR STATION UNIT #1	MADE KG	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.	
ELECTRICAL ELEMENTARY DIAGRAM	CHK'D LGI		
D.C. & MISC.	SQ. CF GC		
	CF. DFN JCG	4192 WORK ORDER	SS-209-024 SIZE DRAWING / 4 REV.
	ENG. W.F. Salter 4-8-71		
	REV. CH. APP DATE 1-KG-FB-W.F.J. 6-21-71	2-KG FB-W.F.J. 10-1-71	3-DP-FB-DS 4-12-73
		4 SEE DCR 2.1318 As Built	

A

CONSTRUCTION
BIDDING PURPOSES ONLY
RELEASED FOR
ENGR.
DATE

C.F. TANK A, MAKEUP
ISOLATION VALVE - CF-V19A

C.F. TANK B, MAKEUP
ISOLATION VALVE - CF-V19B



1) Motor operated from console center. (PB)
2) Both powered from 1C ESV MCC.
3) Opened between 650 psig and 700 psig increasing and breakers EST tagged.
4) Closed at 700 psig decreasing.
5) Located in RB by respective tank.

b. CF-V-2A/B (Sample Line Valves).

1) Motor operated from console center. (PB)
2) Both powered from 1A ESV MCC.
3) Close on ES signal (4 psig & RTI)
4) Located in RB by respective tank.
5) Radiation & HELB Qualified

c. CF-V-3A/B (N_2 Vent Valves)

1) Motor operated from console center. (PB)
2) Powered from A and B RW MCCs.
3) Normally closed with breakers open and EST tagged.
4) Located in RB by respective tank.

d. CF-V-19A/B (N_2/CA Fill Valves)

1) Pneumatically controlled from PL. (PB)
2) Powered from XCR. Fails closed on loss of DC or I.A.
3) Close on ES signal (4 psig & RTI).
4) Located outside RB by RB purge exhaust in Auxiliary Building.
5) Radiation & HELB Qualified

e. CF-V-20A/B (Sample Line).

Content/Skills	Activities/Notes
1) Pneumatically operated from PL. (PB)	
2) Powered from XCL. Fails closed on loss of DC or I.A.	
3) Close on ES signal (4 psig & RTI).	
4) Located outside RB by RB purge exhaust in Auxiliary Building.	
5) Radiation & HELB Qualified	
4. Shroud Heating System	
a. No longer be used.	
b. Low temperature computer alarm based upon setpoint of 70°F; normal shroud temperature is expected to be greater than 90°F due to RB heat.	
. Core Flood System Operation	
1. Normal Parameters	
a. Tank levels 11.35 to 11.81 feet.	
b. Tank pressure 590 to 610 psig.	
c. Tank metal temperature >90°F but <140°F.	
d. Boron concentration 2350 to 2750 ppm.	
1) Upper limit (2850 ppm) based on limiting post-LOCA RB sump solution pH.	
e. Liquid temperature >90°F but <140°F.	
1) Assumed to be same as metal temperature	
2) 140°F is maximum assumed in FSAR.	
2. Core Flood System Maintenance Operations	

Form ES-401-5 **Written Exam Question Worksheet** Q # 007

Examination Outline Cross-Reference Tier # 2

Evolution/System 007 Pressurizer Relief/Quench Tank Group # 1

K/A # K3.01 Page # 3.5-2 RO/SRO Importance Rating 3.3 3.6

Measurement Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 ☐ 55.43

Proposed Question ✓ RO ☐ SRO ✓ PRA Related **Correct Answer** A.

Initial plant conditions:
- PORV leakage to the (RCDT) RC Drain Tank.

Event:
- PORV leakage increases.
- Reactor trip due to low RCS pressure.
- RC Drain Tank Pump WDL-P-8 trips.
- Containment pressure is currently stable at positive 0.2 psig.

Based on these conditions, identify the ONE selection below that describes RCDT overpressure protection if RCDT tank pressure continues to rise.

A. (1) WDG-V-1 Relief Valve opens to relieve RCDT pressure to the REACTOR BUILDING LP Vent Header.
(2) RCDT Rupture Disk bursts to relieve pressure to the RCDT Room.

B. (1) WDG-V-1 Relief Valve opens to relieve RCDT pressure to the AUXILIARY BUILDING LP Vent Header.
(2) RCDT Rupture Disk bursts to relieve pressure to the RCDT Room.

C. (1) WDG-V-1 Relief Valve opens to relieve RCDT pressure to the REACTOR BUILDING LP Vent Header.
(2) RCDT Rupture Disk bursts to relieve pressure to the REACTOR BUILDING LP Vent Header.

D. (1) WDG-V-1 Relief Valve opens relieve RCDT pressure to the AUXILIARY BUILDING LP Vent Header.
(2) RCDT Rupture Disk bursts to relieve pressure to the AUXILIARY BUILDING LP Vent Header.

Technical Reference Lesson Plan 11.2.01.119, Waste Gas Disposal, PPT-38, Rev. 10.
302-694, Waste Gas System, Rev. 43.
OP-TM-220-000, Reactor Coolant System, Section 2.1.26, Page 9, Rev. 4.

Open Exam Reference None.

Learning Objective IV.B.09.01

Question Source ✓ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✓ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer because WDG-V-3 and WDG-V-4 close on RTI signal. This separates the Auxiliary Building and RB LP Vent Headers. Therefore WDG-V-1 discharges to the RB LP Vent header. RCDT Rupture Disk discharges directly to the immediate area in the RC Drain Tank Room (inside the Containment Building).

B INCORRECT because WDG-V-1 discharges to the RB LP Vent Header - which is isolated from the Auxiliary Building LP Vent header due to auto closure of WDG-V-3 and WDG-V-4 (Reactor Trip Isolation).

Distracter is plausible because the rupture disk discharges directly to the immediate area in the RCDT Room, and WDG-V-1 is normally aligned to discharge to the AB vent header.

C INCORRECT because RCDT rupture disk discharges directly to the immediate area in the RCDT Room.

Distracter is plausible because (part 1 is correct) WDG-V-1 discharges to the RB Vent Header under these conditions.

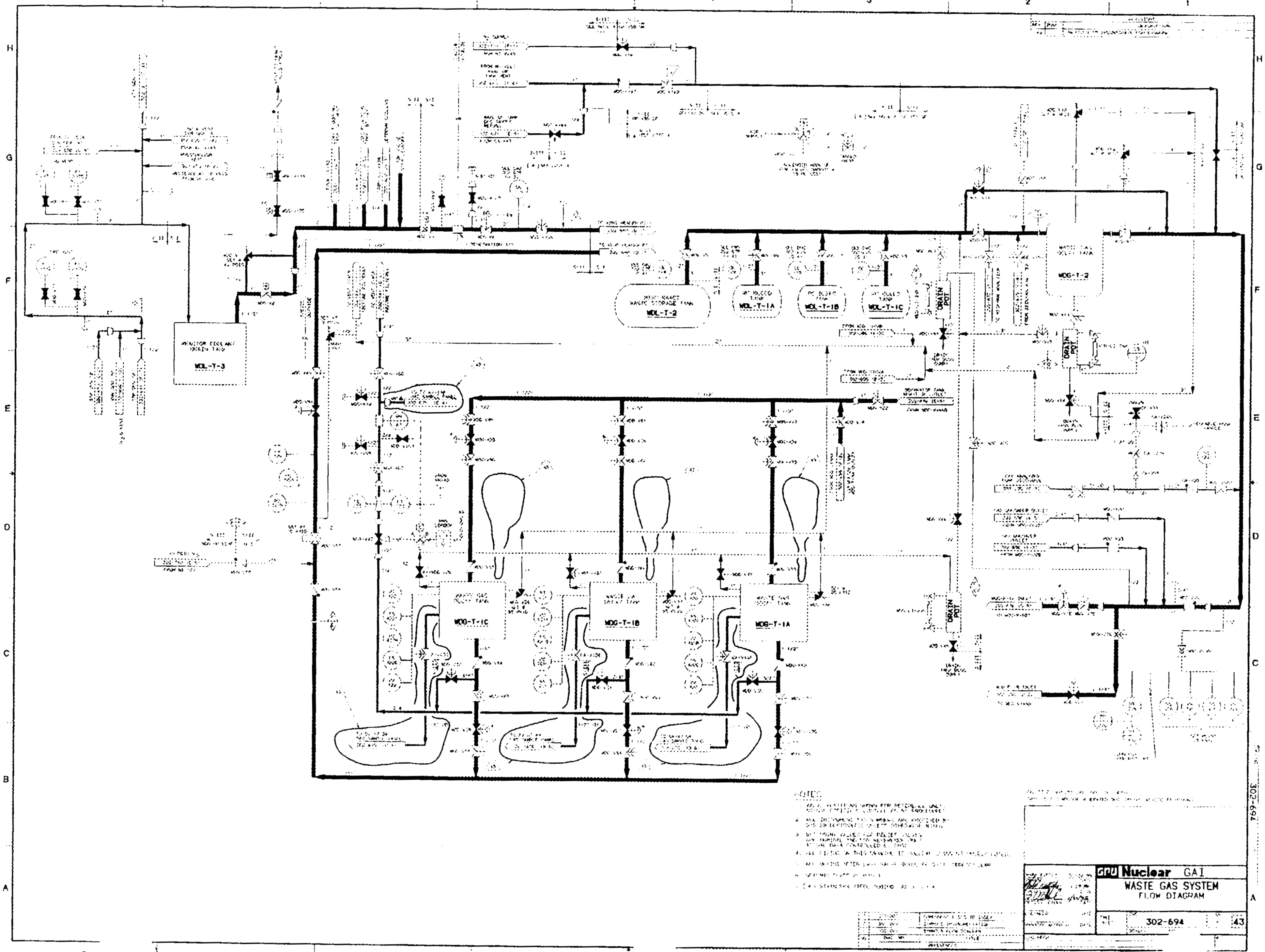
D INCORRECT because WDG-V-1 normally discharges to the Auxiliary Building LP Vent Header.

Distracter is plausible because WDG-V-1 normally discharges to the Auxiliary Building LP Vent Header.

Comments Examinee is required to know:
(1) Sequential overpressure protection for RCDT.
(2) Rupture Disk discharge is to the RCDT room inside containment.
(3) Reactor Trip Isolation closes WDG-V-3 and WDG-V-4, isolating the RB LP Vent Header from the Aux. Bldg LP Vent Header.

WDG-V-3 and WDG-V-4

- Containment Isolation Valves
- 305’ elevation - Inside/Outside RB Wall
- Pneumatic/Motor Operators
- Isolate RCDT and vent header
- **Automatic closure interlocks**
  - **Reactor Trip Isolation**
  - **4# RB Pressure ES Actuation**
  - **RM-G-20 High Radiation Alarm**


REACTOR COOLANT DRAIN TANK
WDL-T-3
WDL-T-2
WDL-T-1A
WDL-T-1B
WDL-T-1C
DRAIN POT
WDG-T-2
WDG-T-1C
WDG-T-1B
WDG-T-1A
NOTES:
GPU Nuclear GAI
WASTE GAS SYSTEM
FLOW DIAGRAM
302-694
43

2.1.15 Maximum and minimum allowable RC pressures during simultaneous operation of RC pumps and the DH system is IAW Figures 1 & 1A of 1102-1 & 1102-11.

2.1.16 If dissolved gases are present in the RC system, the following RCS press & temp limits apply to the operation of CRDS See Figures 1 & 1A of 1102-1 & 1102-11

2.1.17 To maintain valid RCS pressure indication, do **not** vent the pressurizer to the sample line, or attempt to sample the pressurizer, while the low range RC pressure transmitter is in-service.

2.1.18 No action is required in response to LPMS Low alarms (i.e., Sensor Health Warnings on the LPMS Message List) until the plant reaches about 15 percent power.

2.1.19 Boron concentration for the fuel transfer canal during refueling will be maintained at 2450 ppm or greater.

2.1.20 Valves on the fuel transfer tubes (FH-V-1A, FH-V-1B) should **not** be opened until the fuel transfer canal and spent fuel pool "A" are both at the refueling level.

2.1.21 All fuel and/or radioactive material that have been stored in the canal are removed before draining the fuel transfer canal.

2.1.22 The reactor head must be removed prior to flooding the shallow end of the fuel transfer canal in the reactor building.

2.1.23 Do **not** fill the fuel transfer canal through the reactor vessel as it causes a crud burst, which results in high refueling radiation levels. (Except for the LPI test after refueling in accordance with OP-TM-212-211(A) & OP-TM-212-212 (B), LPI Test.)

2.1.24 RCS Degassification should occur prior to cooldown to take advantage of the diluting effect caused by the addition of makeup from RCBT's during cooldown.

2.1.25 If an RCS cooldown was performed without adequate hydrogen degassification (due to emergency cooldown requirements), proceed with degassification IAW OP-TM-220-554. If conditions require immediate opening of RCS before degassification is complete, EXTREME caution should be exercised in the RB to prevent sparks, flames, and other ignition sources from starting a H2 fire until the RB is purged.

2.1.26 The RC drain tank can accept only limited amounts of gas from the pressurizer. Make sure that this tank is **not** overpressurized while venting the pressurizer. RC drain tank relief valve setpoint is 40 psig and RC drain tank rupture disc setpoint is 55 psig.

Form ES-401-5 **Written Exam Question Worksheet** Q # 008

Examination Outline Cross-Reference Tier # 2

Evolution/System 012 Reactor Protection (RPS) Group # 1

K/A # A4.03 Page # 3.7-4 RO/SRO Importance Rating 3.6 3.6

Measurement Ability to manually operate and/or monitor in the control room: Channel blocks and bypasses.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 ☐ 55.43

Proposed Question ✓ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor operating at 100% power, with ICS in full automatic.
- RPS Channel D in MANUAL BYPASS due to an RCS pressure amplifier failure (OFF-SCALE LOW).

Event:
- Monthly surveillance is required to be performed for RPS Channel C.

Based on these conditions identify the ONE selection below that describes the final automatic RPS trip coincidence logic after RPS Channel D is returned to NORMAL with the channel tripped, and Channel C is placed in MANUAL BYPASS.

A. 2 out of 4.

B. 2 out of 3.

C. 1 out of 3.

D. 1 out of 2.

Technical Reference Lesson Plan 11.2.01.132, Reactor Protection System and DSS, Page 13, Rev. 17.

Open Exam Reference None.

Learning Objective IV.E.14.34

Question Source ✓ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A Incorrect because with Channel C in Manual Bypass, trip logic is not 2 out of 4. Channel C will not trip while in manual bypass.

Plausible because the degree of redundancy is one based on Channel A and B available to trip but only 1 channel is required to trip.

B Incorrect because with Channel D already tripped, the trip logic is not 2 out of 3 but 1 out of two.

Plausible because the degree of redundancy is one based on Channel A and B available to trip but only 1 channel is required to trip. Also with Channel C bypassed, the trip logic would be 2 out of 3 if channel D is not already tripped.

C Incorrect because the final trip logic is one out of two.

Plausible because with Channel C in bypass, only 3 channels are available to trip and with Channel D tripped, only one other channel is required to trip.

D Correct Answer. With Channel C in manual bypass and channel D tripped, one out of the remaining two channels is required to trip. With both channels available, the degree of redundancy is one.

Comments In accordance with NUREG 1122, this KA (Ability to manually operate and/or monitor in the control room: Channel blocks and bypasses) is linked to 10CFR55.41(7) (Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features).

The question posed examines the examinee's ability to synthesize control room surveillance activities, status of manually operated RPS channel bypass switches, and channel failure modes in order to expect a reactor trip if ONE operable RPS channel trips. Operational implications of this unusual (allowed by TS) configuration could result in an ATWS with only one channel failure. The stem scenario integrates system knowledges appropriate for an RO level question.

(1) First, two contacts close, applying power to the channel trip relay through an alternate path. With this power applied to the channel trip relay, any trip signal from the RPS channel trip bistables or from the test and critical module relays is over-ridden.

(2) Second, three contacts open preventing the activation of the manual bypass function in any of the other RPS channels. Placing a RPS channel in manual bypass automatically changes the RPS from a 2-out-of-4 logic to a 2-out-of-3 logic for tripping.

3) Indication of manual bypass status is provided by a white light, mounted next to the manual bypass key switch, on the face plate of the reactor trip module.

4) During normal system operations this light will be dimly lit. Whenever the reactor trip module is placed in manual bypass, the light will become brightly lit.

g. Test and Critical Module Relay –

1) The test and critical module relay is powered, through the critical module interlock circuit, from a -15 VDC power supply.

2) Each critical module has an internal jumper within it. These jumpers are all wired in series to form the critical module interlock circuit.

3) When all critical modules are plugged in, the test and critical module relay will be energized, allowing -15 VDC power to pass to the reactor trip relay.

4) Removal of one or more of the critical modules will de-energize the test and critical module relay in the affected channel, thus, de-energizing the RPS channel trip relay.

Examination Outline Cross-Reference Tier # 2

Evolution/System 008 Component Cooling Water Group # 1

K/A # A2.02 Page # 3.8-4 RO/SRO Importance Rating 3.0 3.2

Measurement Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✔ 55.41 .5 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ☐ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor tripped due to loss of offsite power (LOOP).
- MAP C-3-2, IC Surge Tank Level Hi/Lo, actuates.
- IC-T-1 surge tank level is 6 inches, lowering at 1 inch per minute.
- RM-L-9 activity indication is normal and steady.

Based on these conditions, identify the ONE selection below that describes:
(1) Operational impact of this malfunction.
(2) Required actions.

A. (1) Air induction at IC-P-1A/B pump suction can result in slug flow or total loss of IC System flow.
(2) Start Demineralized Water Booster Pump DW-P-1 to establish makeup from Demineralized Water Storage Tank DW-T-1.

B. (1) Air induction into the RM-L-9 counting chamber will result in erroneous activity readings.
(2) Start Demineralized Water Booster Pump DW-P-1 to establish makeup flow from Demineralized Water Storage Tank DW-T-1.

C. (1) Air induction at IC-P-1A/B pump suction can result in slug flow or total loss of IC System flow.
(2) Open IC-V- 5 to establish makeup flow from Demineralized Water Storage Tank DW-T-1.

D. (1) Air induction into the RM-L-9 counting chamber will result in erroneous activity readings.
(2) Open IC-V- 5 to establish makeup flow from the Reclaimed Water Pressure Tank CA-T-7.

Technical Reference 1202-17, Loss of Intermediate Cooling System, Symptom 1.5, Page 2, Rev. 20.
OP-TM-MAP-C0302, IC Surge Tank Level Hi/Lo, Rev 0.
OP-TM-541-463, IC-T-1 Level Control, Section 4.0, Page 2, Rev. 0.

Open Exam Reference None.

Learning Objective IV.B.3.12

Question Source ✔ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because part (2) is in error. Operator is required to open IC-V-5 to establish makeup flow from

DW-T-1.

Distracter is credible because part (1) is correct, part (2) would be the correct answer if this was NS-T-1, the makeup source tank DW-T-1 is correct.

B INCORRECT because part (2) action is wrong (DW-P-1 start is not required - IC-V-5 is required to be opened).

Distracter is plausible because part (1) RMS readings would be affected by air induction into the counting chamber, part (2) source of makeup water for the ICCW System is DW-T-1.

C CORRECT answer.

D INCORRECT answer because part (2) water source (Reclaimed Water Pressure tank CA-T-7) is not correct.

Distracter is plausible because part (1) RMS readings would be affected by air induction into the counting chamber, part (2) makeup water is routed through IC-V-5.

Comments None.

1.0 **SYMPTOMS**

1. I.C. Pump Disch Press Lo, Alarm, 70 psig. (C-2-4)
2. I.C. System Flow Lo, Alarm, 550 GPM. (C-2-2)
3. I.C. CRD Clg Flow Lo, Alarm, 100 GPM. (C-1-2)
4. I.C. CRD Clg Outlet Temp. Hi, Alarm, 160°F. (C-1-3)
5. I.C. Surge Tank Level Hi/Lo, Alarm Hi 24"; Lo 8" (C-3-2) IC-LS-802 or 803
6. I.C. Surge Tank Level "A" Hi/Lo Alarm, Hi 24"; Lo 12"; Lo-2 8". (Computer Pt A0451)
7. I.C. Surge Tank Level "B" Lo, Alarm 8". (Computer Pt A0452)
8. I.C. Cooler Outlet Temp. Hi, Alarm, 120°F. (C-2-3)
9. CRD Stator Temp. Hi, 160°F. Computer Point Area 10 Groups (31-38)
10. I.C. R.C. Pump Cooling Outlet Temp. Hi. (Computer Points A0490, A0491, A0492 and A0493) Setpoint 140°F.

2.0 **IMMEDIATE ACTIONS**

2.1 Automatic Actions

- Standby IC Pump starts (ICCW flow less than 550 GPM)
- MU-V-1A/1B closes (CRD Coolant Outlet Hi Temp. greater than 160°F)

2.2 Manual Action

2.2.1 IF low flow exists, THEN **PERFORM** the following:

2.2.1.1 **VERIFY** OR **START** the standby IC pump.

2.2.1.2 **MONITOR** Surge Tank Level.

2.2.1.3 **FILL** Surge Tank Level as necessary to maintain a normal indicated level of 18.5".

Level 2 – Reference Use

1.0 SETPOINTS

– Hi ≥ 24" from either instrument LSH 802 or LSH 803

– Lo ≤ 8" from either instrument LSL 802 or LSL 803

2.0 CAUSES

Hi

– Leak into IC system from RC pump/letdown coolers

– Demin water makeup valve IC-V-5/open/malfunction

Lo

– System piping or component leak (IC-C-1A/B)

3.0 AUTOMATIC ACTIONS

If level is low and ES actuates, the following occur:

– Channel "A" H.P.I. and Lo level Closes IC-V-3, 4, 6.

– Channel "B" H.P.I. and Lo level Closes IC-V-2, 4, 6.

4.0 MANUAL ACTIONS REQUIRED

4.1 **If** level is high,
then PERFORM the following:

4.1.1 **ENSURE** IC-V-5 is Closed (makeup valve).

4.1.2 **CONTROL** IC-T-1 level (drain) per OP-TM-541-563, IC-T-1 Level Control.

4.1.3 **If** RM-L-9 is rising,
then PERFORM the following:

1. **IAAT** IC-T-1 level indication **cannot** be maintained on scale,
then PERFORM the following:

   A. **TRIP** the reactor.

   B. **TRIP** all four RC pumps.

   C. **CLOSE** IC-V-2 **and** IC-V-3.

2. **COMPARE** trends of the following IC temperatures to determine source of leakage into ICCW:
   - A0490 IC TEMP OUT RC-P-1A CLR
   - A0491 IC TEMP OUT RC-P-1B CLR
   - A0492 IC TEMP OUT RC-P-1C CLR
   - A0493 IC TEMP OUT RC-P-1D CLR
   - A0495 IC TEMP OUT LETDOWN CLR A
   - A0496 IC TEMP OUT LETDOWN CLR B

3. **If** RCP Thermal Barrier is leaking,
   then PERFORM the following:

   A. **CLOSE** breaker associated with isolation valve.
      - IC-V-79A (1A ES Valves MCC, Unit 3D)
      - IC-V-79B (1B ES Valves MCC, Unit 3D)
      - IC-V-79C (1A ES Valves MCC, Unit 7B)
      - IC-V-79D (1B ES Valves MCC, Unit 5A)

   B. **CLOSE** IC-V-79 valve associated with affected RCP.

   C. **BRIEF** all licensed operators on the following requirement:
      - **IAAT** seal injection flow is < 22 gpm,
        then REDUCE Rx power within limits **and STOP** affected RCP IAW OP-TM-226-150 series procedures.

4. **If** Letdown Cooler is leaking,
   then REMOVE affected cooler per one of the following:
   - OP-TM-211-435, Removing and Returning MU-C-1A To Service
   - OP-TM-211-436, Removing and Returning MU-C-1B To Service

4.2 **If** level is low,
then PERFORM the following:

4.2.1 **If** a valid 1600 # ESAS has occurred,
then PERFORM the following:

1. **ENSURE** IC-V-2, 3, 4, and 6 are Closed.

2. **PLACE** IC-P-1A and 1B in PTL.

4.2.2 **If** ESAS HPI Channel A/B has **not** actuated,
then PERFORM the following:

1. **MAINTAIN** surge tank level IAW OP-TM-541-463, IC-T-1 Level Control.

2. **CHECK** system piping/components to determine leak location.

3. **If** surge tank level cannot be maintained > 8",
then INITIATE 1202-17, Loss of ICCW.

4.0 <u>MAIN BODY</u>

NOTE: Normal level is 18.5".

4.1 **OPEN** IC-V-5.

4.2 **When** IC-T-1 is at desired level,
then CLOSE IC-V-5.

5.0 <u>RETURN TO NORMAL</u> - None

6.0 <u>REFERENCES</u>

6.1 <u>Developmental References</u>

6.1.1 1104-8, Intermediate Cooling System, Section 3.2.2.B (superceded)

6.2 <u>Implementing References</u>

6.2.1 OP-TM-541-000, Primary Component Cooling

6.3 <u>Commitments</u> - None

7.0 <u>ATTACHMENTS</u>

7.1 Device Locator List

Form ES-401-5 **Written Exam Question Worksheet** Q # 010

Examination Outline Cross-Reference Tier # 2

Evolution/System 010 Pressurizer Pressure Control System Group # 1

K/A # K5.02 Page # 3.3-7 RO/SRO Importance Rating 2.6 3.0

Measurement Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: Constant enthalpy expansion through a valve.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Initial plant conditions:
- Reactor tripped.
- RCS pressure 1985 psig.
- Pressurizer Relief Valve RC-RV-1A leakage to the RC Drain Tank.
- RC Drain Tank pressure rising.

Event:
- RC Drain Tank relief valve WDG -V-1 fails to open.
- RCDT pressure continues to rise.
  - Rupture disc bursts at 55 psig, reducing RCDT pressure to 5 psig.

Based on these conditions identify the ONE selection below that describes the change in fluid conditions downstream of the Pressurizer relief valve when the RC Drain Tank rupture disc bursts:
(1) Initial tailpipe conditions.
(2) Final tailpipe conditions.

A. (1) 636 degrees F.
(2) 636 degrees F.

B. (1) 355 degrees F.
(2) 245 degrees F.

C. (1) 303 degrees F.
(2) 228 degrees F.

D. (1) 287 degrees F.
(2) 162 degrees F.

Technical Reference Steam Table Book.

Open Exam Reference Steam Table Book.

Learning Objective III.C.02.12

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because this answer is based on the notion that there is no temperature drop during a constant enthalpy expansion process.

Distracter is plausible because it is based on constant temperature, rather than constant enthalpy, and uses Pressurizer saturation temperature based on the stem conditions.

Note:
636 degrees = saturation temperature for 1985 psig.

B INCORRECT because this answer was obtained on the Mollier Diagram by drawing lines from left to right at 1138 BTU/lbm to the 70 and 20 psia curves, and then rising straight up the entropy line up to the saturation dome to obtain erroneous temperatures for 70 and 20 psia saturated steam.

Distracter is plausible, based on an improper method to determine temperature of saturated steam that has moisture content (point is below the saturation dome).

C CORRECT answer.
(1) 303 degrees F is saturation temperature for 55 psig. Initial vapor enthalpy at 1138 BTU/lbm (corresponding to 2000 psia in Pressurizer) expanding down to 55 psig produces wet vapor at 303 degrees F with 4.7% moisture.
(2) 228 degrees F is saturation temperature for 5 psig. Initial vapor enthalpy at 1138 BTU/lbm (corresponding to 2000 psia in Pressurizer) expanding down to 5 psig produces wet vapor with 1.5% moisture.

D INCORRECT because these answers are based on 55 psia and 5 psia, rather than using 55 psig and 5 psig.

Distracter is plausible because it provides correct answers for absolute pressures as listed and plotted in the Steam Table Book.

Note:
162 degrees = 5 psia saturation temperature.
287 degrees = 55 psia saturation temperature.

Comments This question addresses the operational implications of the constant enthalpy expansion through the Pressurizer PORV. The reduction in tailpipe temperature due to downstream pressure reduction (when the rupture disk fails) could lead an operator to believe relief valve leakage has been terminated. This event (RCDT rupture disk failure) actually occurred, and was mis-interpreted by the operators at the start of the 1979 TMI-2 accident.

Form ES-401-5	**Written Exam Question Worksheet**	Q # 011

Examination Outline Cross-Reference Tier # 2

Evolution/System 012 Reactor Protection System Group # 1

K/A # K4.09 Page # 3.7-2 RO/SRO Importance Rating 2.8 3.1

Measurement Knowledge of RPS design feature(s) and/or interlock(s): Separation of control and protection circuits.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 ☐ 55.43

Proposed Question ✓ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

From the list below identify the ONE statement that describes the design feature installed so the SAFETY GRADE Reactor Protection System can supply reactor power, flow and pressure signals to NON-SAFETY GRADE ICS/NNI control systems.

A. Electrical separation of the RPS channels ensures a power supply failure in the control system will not disable more than one RPS channel.

B. Buffer amplifier signal isolation prevents control system failures from feeding back into the RPS and preventing the RPS channel from tripping.

C. Use of independent ICS Calibration Test Modules inside the RPS cabinets ensures control system testing inside the RPS cabinet will not prevent the RPS channel from tripping.

D. Physical separation of the RPS cabinets prevents physical damage (for instance, a control system amplifier fire inside the RPS cabinet) from disabling more than one RPS channel.

Technical Reference SDBD-TI-641, System Design Basis Document for Reactor Protection System (System #641), Pages 3-5 and 3-6, Rev. 3, dated 2/13/2004.

Open Exam Reference None.

Learning Objective IV.E.14.02

Question Source ✓ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✓ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because electrical separation relates to IEEE requirements for channel independence, isolation and separation, rather than isolation from non-safety grade control systems.

Distracter is plausible because electrical separation between channels is an IEEE requirement for the RPS system to reduce the likelihood of interactions between channels during maintenance operations or in the event of a channel malfunction.

B CORRECT answer. All RPS outputs maintain proper independence, isolation, and separation.

C INCORRECT answer because testing of control system signals within an RPS cabinet does not satisfy requirement to isolate signals between safety grade and non-safety grade systems.

Distracter is plausible because testing of the ICS/NNI would not prevent trip of the associated RPS channel.

D INCORRECT answer because physical separation relates to IEEE requirements for channel independence, isolation and separation, rather than isolation from non-safety grade control systems.

Distracter is plausible because physical separation between channels is an IEEE requirement for the RPS system to reduce the likelihood of interactions between channels during maintenance operations or in the event of a channel malfunction.

Comments The RPS neutron flux, RC flow, and RC pressure signals interface with the ICS/NNI systems. These signals are isolated in the RPS before being transmitted to ensure that a failure or failures in the ICS/NNI system components or channel cannot be fed back into the RPS and prevent the protection channel from performing its intended function. If one of the above input signals to the RPS fails, that RPS channel may not respond properly to a plant transient. However, the remaining three RPS channels are still available to provide any required automatic actions.

Design Features: The RPS was designed and fabricated so that channel integrity is maintained under extremes of conditions. Verification of proper system operation under extremes of environmental, including accident, conditions were addressed during equipment qualification. The RPS will operate properly over a range of input energy conditions as described in Section 3.6 of this document. Extremes of energy conditions, e.g., system overvoltage or loss of system power, will cause the RPS to trip. The system can withstand a single failure or malfunction and still perform its protective function (Reference 4.2.13).

3.1.4.6 Channel Independence

Requirement: Channels that provide signals for the same plant protective function shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electric transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction. (IEEE 279-68, Paragraph 4.6)

Design Features: Each RPS channel is located in its own cabinets. The cabinets act as barriers against fire and mechanical damage from external sources. Therefore, physical damage (for instance, an internal fire) can only disable one of the four RPS channels.

Each redundant RPS channel has an independent power source and independent sensors, including independent sensing lines, thermowells and neutron detector wells, which are physically and electrically separated from the other RPS channels and from non-safety systems. All RPS outputs maintain proper independence, isolation, and separation.

Minimal interchannel communication is required for the RPS to perform its protective action. This communication, the 2-out-of-4 reactor trip logic, occurs in the reactor trip module. Relays in the electronic modules of the RPS provide the required isolation between the subsystems even though signals from the various subsystems are combined in the reactor trip module (Reference 4.2.13).

3.1.4.7 Control and Protection System Interaction

Requirement: Where a plant condition that requires protective action can be brought on by a failure or malfunction of the control system, and the same failure or malfunction prevents proper action of a protection system channel or channels designed to protect against the resultant unsafe condition, the remaining portions of the protection system shall independently meet the requirements of paragraphs 4.1 and 4.2 [of IEEE 279-68]. (IEEE 279-68, Paragraph 4.7)

Design Features: The RPS neutron flux, RC flow, and RC pressure signals interface with the ICS/NNI systems. These signals are isolated in the RPS before being transmitted to ensure that a failure or failures in the ICS/NNI system components or channel cannot be fed back into the RPS and prevent the protection channel from performing its intended function.

If one of the above input signals to the RPS fails, that RPS channel may not respond properly to a plant transient. However, the remaining three RPS channels are still available to provide any required protective actions.

3.1.4.8 Derivation of System Inputs

Requirement: To the extent feasible and practical, protection system inputs shall be derived from signals which are direct measures of the desired variables. (IEEE 279-68, Paragraph 4.8)

Design Features: Protection system inputs used in the Safety Analysis for plant protection are derived from signals that are direct measurements of the desired variables. The RPS inputs used by the Safety Analysis to prevent violation of Safety Limits are neutron flux, RC pressure, RC flow, and RC pump status. These inputs are directly measured for actual setpoint comparisons or for calculation of Safety Limit setpoints. Refer to Section 3.3 of this document for signal selection details.

3.1.4.9 Capability for Sensor Checks

Requirement: Means shall be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation.

This may be accomplished in various ways, for example:

(a) by perturbing the monitored variable; or

(b) within the constraints of paragraph 4.11 [of IEEE 279-68], by introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or

(c) by cross checking between channels that bear a known relationship to each other and that have read-outs available.

(IEEE 279-68, Paragraph 4.9)

Design Features: Each RPS channel provides readouts for each monitored parameter. These readouts allow the technician to check sensors by one or more of the following methods:

1. Monitoring the variable after it is perturbed.

2. Use of a substitute input to the sensor of the same nature as the measured variable.

3. Cross-checking the same variable in different channels or other systems.

Form ES-401-5 **Written Exam Question Worksheet** Q # 012

Examination Outline Cross-Reference Tier # 2

Evolution/System 013 Engineered Safety Features Actuation System (ESFAS Group # 1

K/A # K4.13 Page # 3.2-25 RO/SRO Importance Rating 3.7 3.9

Measurement Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following MFW isolation/reset.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 55.43

Proposed Question ✓ RO SRO ✓ PRA Related **Correct Answer** B.

Initial plant conditions:
- Turbine trip from 20% power.
- Automatic EFW actuation due to OTSG 1B low level.
- MFW isolation actuation due to OTSG 1B low pressure.

Current plant conditions:
- OTSG 1A/1B pressures 895 psig.
- OTSG 1B level is being controlled by EFW at the REMOTE AUTOMATIC setpoint.
- Reactor power is 3%.

Based on these conditions, identify the ONE selection below that describes MINIMUM actions required to clear OTSG 1B Low Pressure Feedwater Isolation signal to enable the operator to open FW-V-92B.

A. Press (BOTH) Train A and Train B DEFEAT pushbutton.

B. Press Train B (ONLY) DEFEAT pushbutton.

C. Press (BOTH) Train A and Train B ENABLE pushbuttons.

D. Press 1B Train B (ONLY) ENABLE pushbutton.

Technical Reference Lesson Plan 11.2.01.311, Heat Sink Protection System, Page 13, Rev. 15.

Open Exam Reference None.

Learning Objective IV.E.05.09

Question Source New Bank Question #

✓ Modified Bank Parent Question # QR4E05-06-Q02

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because Train A pushbutton is not required to be operated.

Distracter is plausible because these actions would enable FW-V-92B to be opened - but pressing Train A pushbutton is not required.

B CORRECT answer. Pressing the defeat pushbutton clears the actuation under these conditions.

C INCORRECT answer because pressing the ENABLE pushbuttons above 750 psig has no effect.

Distracter is plausible because the actuation is maintained when OTSG pressure is higher than the (low) pressure actuation setpoint, and it supports the notion that both trains have to be cleared (normal), rather than only one related to a specific component.

D INCORRECT answer because pressing an ENABLE pushbutton above 750 psig has no effect.

Distracter is plausible because the actuation is maintained when OTSG pressure is higher than the (low) pressure actuation setpoint, and it supports the design that only one train needs to be cleared for control of a specific component.

Comments None.

2) system has actuated.

f. Defeat & Enable pushbuttons for Low Press. MFW isolation are grouped together by OTSG. That is Train A & Train B pushbuttons for A OTSG are on CL and the buttons for OTSG - B are on CC.

g. These pushbuttons are backlit to help the operator confirm that the system has responded to his commands.

h. To defeat low pressure isolation BOTH train defeat pushbuttons must be pressed for the desired OTSG.

i. If the enable pushbutton is pressed below 600 psig the system will actuate.

j. If pressure has risen to greater than 750 psig with system actuated. Pressing DEFEAT will remove the actuation and immediately ENABLE the associated train.

6. Bypass switches & Test switches

a. If an instrument must be removed from service due to an instrument failure.

H. Operation

1. Review EF-V-30 operation.

a. Auto operation

1) Air to open, spring to close. Must have handwheel backed all the way out to allow free movement of valve.

b. Manual local operation

1) Counter clockwise turn on the handwheel to open valve.

2. ICS Operating Range & EF-V-30 Control

PPT 38 "A" OTSG Operating range level Optional PPT 39 "B" OTSG Operating Range Level

a. 4 level transmitters, LT 1040, 1041, 1044, 1045

1) Foxboro force balance with diaphragm sensors.

OBJ. 5.9

Examination Outline Cross-Reference Tier # 2

Evolution/System 022 Containment Cooling System (CCS) Group # 1

K/A # A2.06 Page # 3.5-6 RO/SRO Importance Rating 2.8 3.2

Measurement 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: Loss of CCS pump.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✓ 55.41 .5 ✓ 55.43 .5

Proposed Question ✓ RO SRO ✓ PRA Related **Correct Answer** C.

Initial conditions:
- Reactor tripped due to extended loss of off-site power (LOOP).
- Emergency Feedwater suction supplied by the Reactor River Water System.
- RCS at Hot Shutdown conditions.

Event 1:
- RCS LOCA.
- Train A and B ES actuations:
  - 1600 psig, 500 psig and 4 psig RB pressure.
- Reactor Building Spray actuation – BOTH BS-P-1A and BS-P-1B start.

Event 2:
- RB Emergency Cooling Pump RR-P-1A trips – will not restart.

Current conditions:
- RB pressure 32 psig rising slowly.
- All 3 RB Emergency Cooling Units (AH-E-1A, 1B and 1C) operating.

Based on these conditions identify the ONE selection below that describes:
(1) Concern(s) to be addressed by operator action(s).
(2) Action(s) that comply with RR system limits and precautions.

A. (1) Excessive emergency cooling coil differential temperature (cooling water in vs. cooling water out).
(2) Stop RB Cooling Fan AH-E-1A;
Close emergency cooling outlet valve RR-V-4A.

B. (1) Emergency Diesel Generator overload and emergency cooling coil failure.
(2) Stop RB Cooling Fan AH-E-1B;
Close emergency cooling inlet AND outlet valves RR-V-3B and RR-V-4B.

C. (1) Higher (peak) containment temperatures and pressures, and EFW suction supply.
(2) Continue to operate all 3 RB Cooling Fans;
Maintain Emergency Feedwater supply valves EF-V-4 and EF-V-5 OPEN.

D. (1) RR-P-1B pump runout, and prioritization of containment cooling over EFW during LOCA conditions.
(2) Continue to operate all 3 RB Cooling Fans;
Close Emergency Feedwater supply valve EF-V-4 or EF-V-5.

Technical Reference 1104-38, Reactor Building Emergency Cooling Water System, Limit & Precaution 2.2.11, Page 6, Rev. 57.
OP-TM-534-901, RB Emergency Cooling Operations, Step 3.2.1 and the Caution before Step 4.1.5, Pages 1 and 2, Rev. 3.

Written Exam Question Worksheet

Open Exam Reference None.

Learning Objective IV.A.17.14

Question Source ✔ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because no procedures guide the operator to perform these actions, and there is no concern for cooling water delta-T.

Distracter is plausible because other plant coolers have Delta-T design limits, and the action described would reduce Delta-T.

B INCORRECT because Diesel Generator load limit is not referenced in the 1104-38 system limits and precautions for this condition. Also, incorrect actions are described.

Distracter is plausible because Emergency Diesel Generator loading limits have small margin of safety, and actions described would reduce flow (load).

C CORRECT answer.

D INCORRECT because the actions described (close EF-V-4 or EF-V-5) are incorrect - not directed by any procedures.

Distracter is plausible because excessive pump flow is a concern as described in OP-TM-534-901, RB Emergency Cooling Operations, and the described actions would reduce RR-P-1B flow. OP-TM-534-901 does not state whether motor load (runout) is the actual concern.

Comments None.

6. The dp across RR-S-1A and B shall be less than 6 psid. If dp is greater than or equal to 6 psid, then declare the affected train inoperable. If any either strainer is inoperable, enter TS 3.3.2.

   ❶ Strainer operation does **not** affect train operability provided that strainer dp can be maintained less than 6 psid. However, all strainer problems shall be addressed promptly to ensure strainer dp is < 6 psid.

   ❷ Closing the strainer backwash valves, removes the ability of the operating strainers to clean themselves. During strainer maintenance, it is acceptable to close **all** of the strainer backwash valves (to prevent back flow) only if the river is clean. The time period during which all the backwash valves are closed should be minimized. Suggest that the maintenance activity blank off the strainer backwash header on the strainer to be worked.

   ❸ Enclosure 7 provides guidance for hand rotation of the strainer and guidance for manual operation for the strainer backwash valve.

7. During emergency standby, if the strainer backwash valve(s), RR-V-33 A/B is to be closed, RR-V-5 shall be opened to prevent deadheading the RR Pump(s) if RR-V-6 fails closed (re. CR 145653). See Section 3.6 of this procedure.

8. Maintain **single** RR Pump flow (as indicated by total of Computer Points A1049, A1050 and A1051) less than or equal to 5800 gpm if ISPH pump bay water level (as indicated by SR-LI-1172) is > 277' or less than or equal to 5600 gpm if ISPH pump bay water level is ≤ 277'. This assures adequate pump NPSH is achieved, especially if RR-V6 fails open or if RR-V5 is throttled open to control backpressure. Normal ISPH pump bay water level is > 277'. (Re. C-1101-534-E410-020, Rev. 1).

   a. Upon loss of all control air RR-V-6 will not fully fail open because in 13R, 1303-11.9 short stroked RR-V-6 to 5800 gpm (with 1 RR pump running). 5800 gpm equates to 64 degrees open, as indicated by the local valve stem scale.

9. EFV-4 and EFV-5 (River Water Isolation to EFP suction) will be checked closed and their associated breakers on 1C E.S. Valve Motor Control Center will be locked open to prevent inadvertent admission of river water to the OTSG's.

10. The blank flange between EF-V-4 and EF-V-5 will be checked in the blanked position.

11. If Reactor Building emergency cooling is required and only one (1) RR pump is operable:

    a. Operate all available coolers that have AH-E-1 fans in service. Operate fans in SLOW speed if a RB 4 psig actuation has occurred.

    b. Isolate any cooler that does not have AH-E-1 fan(s) in service, by closing the associated RR-V-4(s).

Form ES-401-5 **Written Exam Question Worksheet** Q # 014

Examination Outline Cross-Reference Tier # 2

Evolution/System 022 Containment Cooling System (CCS) Group # 1

K/A # A1.02 Page # 3.5-6 RO/SRO Importance Rating 3.6 3.8

Measurement Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✔ 55.41 .5 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ✔ PRA Related **Correct Answer** D.

Plant conditions:

- Design basis LOCA has occurred.
- Both Reactor Building Spray Pump breakers are tripped.
- Reactor Building Ventilation Recirculation Unit AH-E-1A is running in SLOW with NO Reactor River water cooling flow.
- Reactor Building Ventilation Recirculation Units AH-E-1B and AH-E-1C are running in SLOW with full Reactor River cooling water flow.

Based on these conditions and equipment failures, identify the ONE selection that describes required action(s) that satisfy design basis requirements.

A. Stop AH-E-1A.

B. Stop AH-E-1A then re-start it in FAST.

C. One at a time, shift each unit to FAST and establish full cooling water flow.

D. Correct the valve lineup to establish full cooling water flow through AH-E-1A.

Technical Reference Lesson Plan 11.2.01.128, Reactor Building Heating & Ventilation System, Section J Design Basis, Page 11, Rev. 8.

Open Exam Reference None.

Learning Objective IV.F.02.02

Question Source ✔ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✔ Memory/Fundamental Knowledge Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because all 3 fans are required to meet basis operating configuration.

B INCORRECT because fans should not be operated in fast speed during LOCA conditions to prevent motor overload conditions.

C INCORRECT because fans should not be operated in fast speed during LOCA conditions to prevent motor overload conditions.

D CORRECT in accordance with design basis for accident condition operation with 0 Spray Pumps operating.

Comments Fans are operated during LOCA for energy removal to limit peak temperatures and building pressure (note that the building spray pumps are failed in the stem). The parameter addressed is CCS cooling water flow to emergency cooling coils in order to prevent exceeding containment pressure limit.

The correct answer identifies action required to correct a parameter (emergency cooling coil river water flow) associated with operating the CCS in to prevent exceeding Containment pressure design limit (both RB Spray Pumps are tripped in the stem). Operation of the CCS is now the only means of limiting RB pressure to less than the design limits as described in the technical reference for this question.

Limits are to prevent non-ductile failure of valve bodies.

5. Purging operation when Containment Integrity is required requires a 31° (33°) opening restriction on the Purge isolation valves so that they will more easily close under higher D/P if necessary in the event of an accident.

NOTE: This item was an identified weakness for CRD Group 8 licensing class. Ensure it is understood.

   a. Done by mechanical means and limit switches on operators.

   b. No limit when Containment Integrity is not required - valves may be opened the full 90°.

6. R.B. recirc. fan motor current must be monitored during Integrated Leak rate Testing (ILRT) to avoid motor overload.

7. Normal operation requires both Operating Floor Vent fans (AH-E-3A &B) be operated simultaneously.

8. Reactor shall not be made critical unless 2 RB fans are capable of being operated (T.S. 3.3.1.3)

I. Design Basis

The following combinations of RB. Emergency cooling units and Spray Pumps will handle the R.B. pressure and temperature conditions resulting from the worst case design basis accident.

2 Spray Pumps and 0 Emergency Coolers

1 Spray Pump and 1 Emergency Cooler

0 Spray Pumps and 3 Emergency Coolers

The R.B. recirc fan/cooler units have relief valves to protect them from high pressure implosive forces during an accident condition.

1. The Purge Supply and Exhaust valves are designed to close against a pressure 60 PSI and when closed they must seal bubble tight at 125 PSIG.

   Exterior, air operated valves must close fully in 2 seconds.

Examination Outline Cross-Reference Tier # 2

Evolution/System 026 Containment Spray System (CSS) Group # 1

K/A # A3.01 Page # 3.5-12 RO/SRO Importance Rating 4.3 4.5

Measurement Ability to monitor automatic operation of the CSS, including: Pump starts and correct MOV positioning.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 55.43

Proposed Question ✓ RO SRO ✓ PRA Related **Correct Answer** A.

Initial conditions:

- Reactor trip from full power due to LOCA.
- RB pressure reached 5 psig.
- Manual ESAS actuation signals were NOT initiated.
- Automatic ES Actuation status:

	Train A	Train B
1600#	Bypassed	Bypassed
4#	Defeated	Actuated (NOT defeated)

BASED ON THESE INITIAL CONDITIONS, identify the ONE statement below that describes automatic component response if RB pressure rises rapidly (spikes) to 40 psig.

A. Only BS-P-1B starts.

B. Both BS-P-1A and BS-P-1B start.

C. Only BS-P-1A starts;
Train 'A' RB Spray (BS) valves open.

D. Both BS-P-1A and BS-P-1B start;
Both Train 'A' and Train 'B' RB Spray (BS) valves open.

Technical Reference Lesson Plan 11.2.01.127, Reactor Building Spray System, PPT 14 and PPT 26, Rev. 17.

Open Exam Reference None.

Learning Objective IV.E.24.07

Question Source New ✓ Bank Question # NRC-21 Q-062.

Modified Bank Parent Question #

Question NRC Exam History TMI 2003 Q-062

Question Cognitive Level Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. The initial conditions given in the stem show that Block 4 BS start permissive has been disabled for Train A on 4#. Train B BS start permissive is in effect for the 4# actuation.

B INCORRECT answer. BS-P-1A does not start, since Train A actuations are bypassed/defeated (Block 4 start permit does not exist for BS-P-1A).

Distracter is plausible because the 30# ES actuation will cause building spray pumps to start on any train that has a Block 4 permissive signal.

C INCORRECT because with train A 4# actuations defeated, the Block 4 BS permissive start signal is removed. Therefore, BS-P-1A will not start. Also, Train A RB Spray valves opened on the 4# ES as Block 2 loads.

Distracter is plausible because the 30# ES actuation will cause building spray pumps to start on any train that has a Block 4 permissive signal.

D INCORRECT. Train "A" BS pump will not start due to not having a block 4 start signal and Train A and Train B building spray valves are already open.

Distracter is plausible because BS-P-1B will start on the 30# ES signal due to having a Train B block 4 start signal.

Comments 2003 TMI SRO-21 NRC Exam Q-062.

Component Descriptions

BS-P-1A/B

- Operating Modes
  - AUTO
    - Auto start at 30 psig increasing RB pressure (2 out of 3 pressure switches) **Must have Block 4 permissive.**
  - Manual
    - From Control Room Console

System Operation

- Normal Operation (initiated by 2 of 3 RB pressure sensors at 4 and 30 psig)
  - 4 psig valves start to open within 10 seconds
    - DH-V-5A/B
    - BS-V-1A/B
    - BS-V-2A/B
    - BS-V-3A/B
  - 30 psig BS-P-1A/B start
  - As RB pressure drops, flow will increase

Form ES-401-5 **Written Exam Question Worksheet** Q # 016

Examination Outline Cross-Reference Tier # 2

Evolution/System 026 Containment Spray System Group # 1

K/A # K1.01 Page # 3.5-10 RO/SRO Importance Rating 4.2 4.2

Measurement Knowledge of the physical connections and/or cause-effect relationships between CSS and the following systems: ECCS

10 CFR Part 55 Content ✔ 55.41 .2 to .9 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ✔ PRA Related **Correct Answer** B.

Sequence of events:

Time	RCS Pressure	Event
0950	1900 psig	Reactor trip from full power due to LOCA.
1000	1200 psig	HPI systems injecting BWST water into the RCS.
1030	800 psig	RB Pressure peaked at 6 psig.
1130	700 psig	LPI/HPI Pumps in "Piggyback" Mode.
1200	550 psig	Core Flood Tanks dumping.
1230	150 psig	LPI systems injecting BWST water into RCS.

- RB Spray Pumps are NOT operating.
- RB Sump recirculation is NOT initiated.

Based on this event, identify the EARLIEST TIME below when sodium hydroxide (NaOH) will actually be injected into the reactor core.

A. Time = 1000.

B. Time = 1130.

C. Time = 1200.

D. Time = 1230.

Technical Reference 302-640, Decay Heat Removal Flow Diagram, Rev. 79.

Open Exam Reference None.

Learning Objective IV.A.11.02

Question Source ☐ New ✔ Bank Question # NRC-21 Q-061.

☐ Modified Bank Parent Question #

Question NRC Exam History TMI 2003 Q-061

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. NaOH tank ties into LPI suction header, but not HPI suction.

Distracter is plausible because this is the earliest point that ECCS water is entering the RCS and BS-V-2A/B, NaOH isolation valves are open on block loading.

B CORRECT answer. NaOH tank is being pumped to MUP suction by the DH Pumps, even though there is no direct LPI flow into the RCS.

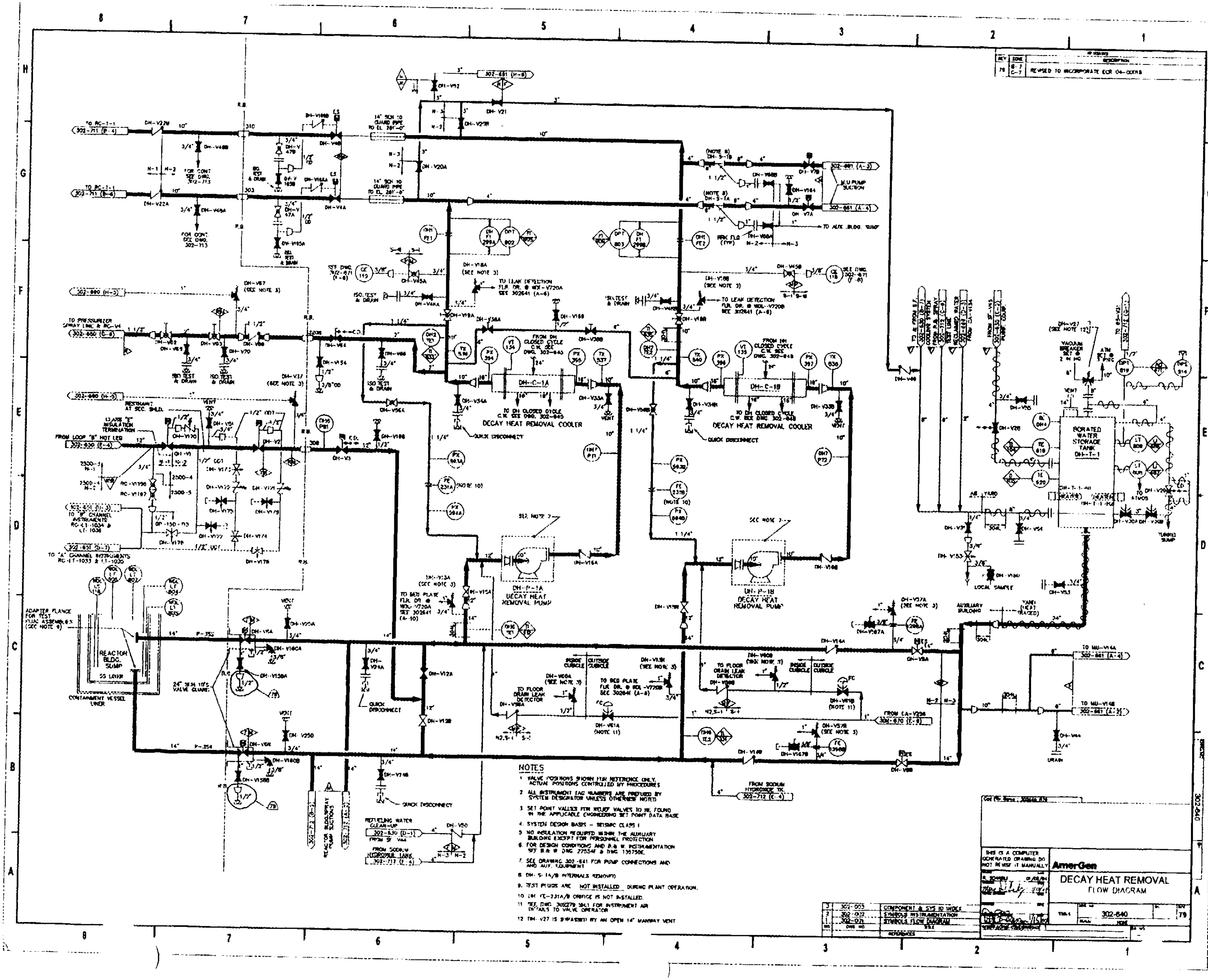
C INCORRECT. NaOH is already being added in a previous answer. Core Flood Tank water does not contain NaOH.

Distracter is plausible because RCS pressure is close to the shutoff head of the LPI pumps.

D INCORRECT. NaOH is already being added in a previous answer. This would provide additional NaOH to the ECCS.

Distracter is plausible because with BS-V-2A/B open during the initial block loading, NaOH is lined up to the suction of the LPI pumps.

Comments 2003 TMI SRO-21 NRC Exam Q-061.



REVISED TO INCORPORATE ECR 04-00018
TO RC-T-1
302-711 (B-4)
302-711 (B-6)
FOR CONT SEE DWG. 302-713
14" SCH 10 GUARD PIPE TO EL 281'-0"
302-661 (H-6)
302-661 (A-3)
302-661 (A-4)
M.U PUMP SUCTION
TO AUX. BLDG. SUMP
TO PRESSURIZER SPRAY LINE & RC-V4
302-650 (E-6)
302-690 (H-3)
302-690 (H-3)
FROM LOOP "B" HOT LEG
302-650 (E-4)
TO LEAK DETECTION FLR. DR. @ WDL-V720A SEE 302641 (A-6)
TO LEAK DETECTION FLR. DR. @ WDL-V720B SEE 302641 (A-6)
FROM DH CLOSED CYCLE C.W. SEE DWG. 302-645
TO DH CLOSED CYCLE C.W. SEE DWG. 302-645
DH-C-1A
DH-C-1B
DECAY HEAT REMOVAL COOLER
QUICK DISCONNECT
DH-P-1A
DECAY HEAT REMOVAL PUMP
DH-P-1B
DECAY HEAT REMOVAL PUMP
BORATED WATER STORAGE TANK DH-T-1
LOCAL SAMPLE
AUXILIARY BUILDING
REACTOR BLDG. SUMP
CONTAINMENT VESSEL LINER
24" SCH 10'S VALVE GUARD
ADAPTER FLANGE FOR TEST PLUG ASSEMBLIES (SEE NOTE 9)
INSIDE CUBICLE
OUTSIDE CUBICLE
TO FLOOR DRAIN LEAK DETECTOR
TO BED PLATE FLR. DR. @ WDL-V720B SEE 302641 (A-6)
FROM CA-V256
302-670 (E-8)
TO MU-V14A
302-661 (A-4)
TO MU-V14B
302-661 (A-3)
DRAIN
FROM SODIUM HYDROXIDE TK.
302-712 (E-4)
REFUELING WATER CLEAN-UP
302-630 (D-1)
FROM SF V44
REACTOR BLDG. SPRAY PUMP SUCTIONS
302-712 (B-2)
302-712 (A-2)

NOTES
1. VALVE POSITIONS SHOWN FOR REFERENCE ONLY. ACTUAL POSITIONS CONTROLLED BY PROCEDURES
2. ALL INSTRUMENT TAG NUMBERS ARE PREFIXED BY SYSTEM DESIGNATOR UNLESS OTHERWISE NOTED
3. SET POINT VALUES FOR RELIEF VALVES TO BE FOUND IN THE APPLICABLE ENGINEERING SET POINT DATA BASE
4. SYSTEM DESIGN BASIS – SEISMIC CLASS I
5. NO INSULATION REQUIRED WITHIN THE AUXILIARY BUILDING EXCEPT FOR PERSONNEL PROTECTION
6. FOR DESIGN CONDITIONS AND B.& W. INSTRUMENTATION SEE B.& W. DWG. 27554F & DWG. 135790E
7. SEE DRAWING 302-641 FOR PUMP CONNECTIONS AND AND AUX. EQUIPMENT
8. DH-S-1A/B INTERNALS REMOVED
9. TEST PLUGS ARE NOT INSTALLED DURING PLANT OPERATION.
10. DH-FE-231A/B ORIFICE IS NOT INSTALLED.
11. SEE DWG. 302279 SH.1 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
12. DH-V27 IS BYPASSED BY AN OPEN 14" MANWAY VENT

3 302-003 COMPONENT & SYS ID INDEX
2 302-002 SYMBOLS INSTRUMENTATION
1 302-001 SYMBOLS FLOW DIAGRAM
NO. DWG NO. TITLE
REFERENCES
THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY
AmerGen
DECAY HEAT REMOVAL
FLOW DIAGRAM
302-640
79

Form ES-401-5 **Written Exam Question Worksheet** Q # 017

Examination Outline Cross-Reference Tier # 2

Evolution/System 039 Main and Reheat Steam System (MRSS) Group # 1

K/A # A2.01 Page # 3.4-21 RO/SRO Importance Rating 3.1 3.2

Measurement Ability to (a) predict the impacts of the following mal-functions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Flow paths of steam during a LOCA.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✔ 55.41 .5 55.43

Proposed Question ✔ RO SRO ✔ PRA Related **Correct Answer** C.

Plant conditions:
- Small break LOCA occurred coincident with a loss of off-site power (LOOP).
- ES Train A and Train B actuated.
- Both EDGs are supplying their respective bus.
- Average of the five highest core exit thermocouples is 556 degrees F.
- RCS Tavg is 546 degrees F.
- RCS cooldown rate is 20 degrees per hour.
- Primary-to secondary heat transfer does NOT exist.
- RCS Pressure stable at 1230 psig.

Event:
- In accordance with EOP-006, LOCA Cooldown, you are directed to "Reduce OTSG pressure so that secondary Tsat is 60 degrees F lower than incore thermocouple temperature."

Based on these conditions identify the ONE correct action below to establish an acceptable RCS cooldown rate.

A. Open the Turbine Bypass Valves to reduce OTSG pressure to 640 psig.

B. Open the Turbine Bypass Valves to reduce OTSG pressure to 743 psig.

C. Open the Atmospheric Dump Valves to reduce OTSG pressure to 640 psig.

D. Open the Atmospheric Dump Valves to reduce OTSG pressure to 743 psig.

Technical Reference Steam Table Book.

Open Exam Reference Steam Table Book.

Learning Objective V.E.18.02

Question Source ✔ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. Correct pressure range but TBVs are not available due to LOOP - condenser is not available.

Distracter is plausible because (640 psig) OTSG pressure is correct for the conditions in the stem.

B INCORRECT. TBVs are not available and this pressure range is based on primary Tsat.

Distracter is plausible since examinee could incorrectly use RCS Tsat to determine OTSG pressures - the basis for these incorrect values. RCS Tsat is 572 degrees, based on 1230 psig.

572 – 60 = 512 degrees, corresponding to 743 psig (OTSG pressure).

C CORRECT. The ADVs are the correct components to control OTSG pressure and the pressure given is correct for 60 degrees F below incore thermocouples.

556 degrees – 60 = 496 degrees, corresponding to 640 psig (OTSG pressure).

D INCORRECT. These are incorrect values based on primary Tsat.

Distracter is plausible because the ADVs are the correct components to control OTSG pressure and the pressures given are correct if the examinee had a misconception and used RCS Tsat to calculate OTSG pressures instead of incore thermocouples.

Comments None.

Form ES-401-5 **Written Exam Question Worksheet** Q # 018

Examination Outline Cross-Reference Tier # 2

Evolution/System 059 Main Feedwater Group # 1

K/A # 2.1.23 Page # 2-3 RO/SRO Importance Rating 3.9 4.0

Measurement 45.2, 45.6
Ability to perform specific system and integrated plant procedures during all modes of plant operation.

10 CFR Part 55 Content 55.41 .... 55.43 ....

Proposed Question ✓ RO SRO PRA Related **Correct Answer** D.

Plant conditions:

- Plant heatup in progress using 1102-1, Plant Heatup To 525 degrees F.
- FW-P-1A is being placed in service using OP-TM-401-103, Shifting FW-P-1A From Standby Mode To Operating Mode.
- FW-P-1A trip function has been tested.

In accordance with OP-TM-401-103, identify the ONE selection below that describes operation of FW-P-1A speed controls, starting with the feedpump on the turning gear.

A. Use ICS FP "A" Turbine Speed Demand control station in HAND. After FW valve dP is 60-90 psid, transfer this control station to AUTO.

B. With the ICS FP "A" Turbine Speed Demand control station in AUTO, adjust the ICS FP A Bias Control until FW valve dP is 60-90 psid.

C. Turn FP "A" Turbine Full Range Manual Speed Control Switch to ON. When the High Speed Stop (HSS) lamp is lit transfer the control switch to OFF.

D. Use 1A FPT Governor control switch until speed stops increasing - then position and hold the control in FAST RAISE until the HSS lamp is lit. When additional speed is required, use ICS FP "A" Turbine Speed Demand control station in HAND.

Technical Reference OP-TM-401-103 Step 4.3.7.2, Page 3, Rev. 2.

Open Exam Reference None.

Learning Objective IV.C.4.07

Question Source ✓ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✓ Memory/Fundamental Knowledge Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the initial speed is below the ability of ICS and the Air Speed Changer Motor to control FWP speed.

Distracter is plausible because once the FWP speed has risen to ICS controllable speeds, this is the correct action.

B INCORRECT because the initial speed is below the ability of ICS and the Air Speed Changer Motor to control FWP speed. Also, the FWP bias control does not regulate valve D/P.

Distracter is plausible because with the FWP in auto, the bias control will affect its load sharing with the second FWP.

C INCORRECT because the full range manual speed control switch is in off per the procedure.

Distracter is plausible because the full range manual speed control will control the FWP turbine through the full range of turbine speeds.

D CORRECT. The governor control switch is used to raise speed until the ICS control station takes over.

Comments None.

4.3.6 **VERIFY** speed control is at initial conditions and **RESET** FW-P-1A Turbine as follows:

1. **VERIFY** 1A FPT Governor at the LSS. ____

2. **VERIFY** FP A Turbine Full Range Manual Speed Control switch Off. ____

3. **VERIFY** ICS FP A Turbine Speed Control in Hand with Zero demand. ____

4. **VERIFY** ICS FP A bias control at 0%. ____

5. **RESET** FW-P-1A by pressing Reset PB (CL) ____

6. **VERIFY** the following:

   - Green reset light Lit ____
   - Red trip light Off ____
   - Annunciator M-1-1 Clear ____
   - Low Pressure Stop Valve (LPSV) and High Pressure Stop Valve (HPSV) indicate Open (red light Lit) ____

4.3.7 **APPLY** steam and **PLACE** Turning Gear in standby as follows:

1. **STATION** an Operator at the FWP A Turbine to observe and listen for any abnormal conditions when steam is first admitted. ____

2. **RAISE** Turbine speed demand with 1A FPT Governor (CL) to the point where sufficient steam is admitted to turn the rotor (FW-SR-6) and/or (A0320). ____

3. **OBSERVE** the MSC handwheel responds to the signal to Raise Turbine Speed.

   - **IF** the handwheel does not respond, **THEN** have the local observer manually turn the handwheel slowly in the "raise" direction (CCW) 1-2 turns.
   - **REPEAT** Step 4.3.7.2 to test response again.

4. **VERIFY** that the Turning Gear is disengaged by the following:

   - Engage light Off ____
   - Disengage light Lit ____

Examination Outline Cross-Reference Tier # 2

Evolution/System 061 Auxiliary/Emergency Feedwater (AFW) System Group # 1

K/A # A2.01 Page # 3.4-47 RO/SRO Importance Rating 2.5 2.6

Measurement Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Startup of MFW pump during AFW operation.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✔ 55.41 .5 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ☐ PRA Related **Correct Answer** A.

Initial plant conditions:
- Reactor operating at 60% power.
- FW-P-1A is the only available Main FW Pump.

Event:
- Reactor trip due to FW-P-1A trip.

Current plant conditions:
- All 3 Emergency Feedwater Pumps are operating.
  - EFW control valves EF-V-30A-D are controlling OTSG levels at setpoint.
- RCS subcooling margin is 41 degrees F and steady.
- FW-P-1A has been restarted, and is in HAND maintaining 0.1 mlbm/hr to each OTSG.

Based on these conditions identify the ONE selection below that describes required disposition of EFW equipment.

A. Return EFW systems to normal standby conditions.

B. Stop all EFW pumps, and manually close EF-V-30A-D.

C. Continue operating EFW as the preferred source of FW.

D. Continue operating EFW as a back up to the only operating MFW pump.

Technical Reference OP-TM-EOP-010, Guide 15.1 Return EFW to Standby, Page 31, Rev. 3.

Open Exam Reference None.

Learning Objective V.E.10.02

Question Source ✔ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. All of the conditions for returning EFW to standby are met, therefore EFW must be returned to standby. In accordance with OS-24, operators are responsible to recognize conditions which apply to EOP-010 Rules and Guides.

B INCORRECT because the prerequisites for EFW shutdown are not met.

Distracter is plausible because with the reactor shutdown and MFW available, there can be a misconception that EFW can be simply shutdown without returning the systems to standby.

C INCORRECT because the conditions for returning EFW to standby are met, therefore EFW must be returned to standby.

Distracter is plausible because EOP-010 Rule 4 identifies EFW as preferred source of FW under other operating conditions.

D INCORRECT because the conditions for returning EFW to standby are met, therefore EFW must be returned to standby.

Distracter is plausible because of perceived risk due to failure vulnerability with only one FW Pump operating.

Comments In this question the examinee is required to evaluate impact of Startup of one MFW pump during AFW operation; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations. The correct answer has the examinee apply procedure knowledge of Guide 15.1 AFW "return to standby criteria" to existing plant conditions, and recognize that AFW operation is no longer required. This same procedure, Guide 15.1, is used to return EFW to standby. Writer purposely did not identify Guide 15.1 title, Return EFW to Standby, for obvious reasons.

Guide 15.1
Return EFW to Standby

When ALL of the following conditions are satisfied,

_____ SCM > 25°F
_____ Main Feedwater flow has been established to each available OTSG
_____ At least one reactor coolant pump is operating
_____ OTSG level > 20" in each available OTSG.
_____ RB pressure < 2 psig
_____ CRS concurrence has been obtained

then PERFORM the following to place EFW in standby.

1. **PLACE** the EFW control valves in Manual

_____	EF-V-30A	_____	EF-V-30B
_____	EF-V-30D	_____	EF-V-30C

_____ 2. **ENSURE** all EFW actuation switches (8) are in DEFEAT.

_____ 3. **CLOSE** EF-V-30A & D **and ENSURE** OTSG A level is maintained with Main FW

_____ 4. **CLOSE** EF-V-30B & C **and ENSURE** OTSG B level is maintained with Main FW

_____ 5. **PLACE** Train A **and** Train B EFW Actuation switches for Loss of RCPs **and** High RB Pressure in ENABLE. (4 switches)

_____ 6. **If** at least one FW pump is RESET, **then PLACE** Train A **and** Train B EFW Actuation for Loss of FWPs in ENABLE (2 switches)

_____ 7. **If** OTSG A level > 20" and OTSG B level > 20", **then PLACE** Train A **and** Train B EFW Actuation for Lo-Lo OTSG Level in ENABLE (2 switches)

_____ 8. **PLACE** EF-P-2A in Normal-after-stop

_____ 9. **PLACE** EF-P-2B in Normal-after-stop

_____ 10. **ENSURE** MS-V-10A is CLOSED **and CLOSE** MS-V-13A

_____ 11. **ENSURE** MS-V-10B is CLOSED **and CLOSE** MS-V-13B

12. **PLACE** each EFW control valve in AUTO **and SELECT** REMOTE setpoint

_____	EF-V-30A	_____	EF-V-30B
_____	EF-V-30D	_____	EF-V-30C

Form ES-401-5 **Written Exam Question Worksheet** Q # 020

Examination Outline Cross-Reference Tier # 2

Evolution/System 061 Auxiliary/Emergency Feedwater (AFW) System Group # 1

K/A # A1.01 Page # 3.4-47 RO/SRO Importance Rating 3.9 4.2

Measurement Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) asociated with operating the AFW controls including: S/G level.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✓ 55.41 .5 55.43

Proposed Question ✓ RO SRO PRA Related **Correct Answer** D.

Plant conditions:
- Reactor tripped due to loss of off-site power (LOOP).
- EF-P-1 has tripped
- All other Emergency Feedwater System Components have responded as designed.
- OTSG levels are at 40% on the Operating Range and increasing
- Emergency Feedwater control valves EF-V-30A, B, C, D in AUTO.
- RCS is subcooled.

Event:
- HSPS switches for Loss of All Reactor Coolant Pumps are taken to DEFEAT in preparation for resetting EF-P-1.

Based on these conditions identify the ONE selection below that describes OTSG level control setpoint after the HSPS switch operation described above.

A. 0% on the Operating Range.

B. 25" on the Startup Range.

C. Current value of 40% on the Operating Range.

D. 50% on the Operating Range.

Technical Reference Lesson Plan 11.2.01.311, Heat Sink Protection System, Page 19, Rev. 15.

Open Exam Reference None.

Learning Objective IV.E.05.02, IV.E.05.07

Question Source New Bank Question #

✓ Modified Bank Parent Question # QR4E05-02-Q01.

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the EF-V-30 valves will continue to control OTSG levels at 50% due to the seal in circuit in HSPS which in effect locks out the Loss of All RCPs switches going to defeat.

Distracter is plausible because defeating the loss of all RCPs input signal removes the EF start signal which, if EFW was reset, would return setpoint control to the EFW not initiated setpoint.

B INCORRECT because the EF-V-30 valves will continue to control OTSG levels at 50% due to the seal in circuit in HSPS. The seal in circuit would prevent EFW control from returning to the 25" Startup Range setpoint.

Distracter is plausible because the only HSPS actuation that causes EF-V-30s to control at 50% is Loss of All

RCPs. Removal of this signal by defeating the loss of all RCPs input signal would return setpoint control to 25" except for the seal in circuit.

C INCORRECT because the EF-V-30 valves will continue to control OTSG levels at 50% due to the seal in circuit in HSPS.

Distracter is plausible because if the EF-V-30s were placed in local setpoint and the setpoint changed to 20%, then EF-V-30s would continue to control at this level.

D CORRECT. The EF-V-30 valves will continue to control OTSG levels at 50% due to the seal-in circuit in HSPS.

Comments Modified TMI Bank Question QR4E05-02-Q01.

Content/Skills	Activities/Notes
5. EF-P Auto start and OTSG level control setpoint selection a. Output of OTSG S/U Range (Type 2 compensated) is sent to two level bistables (A/D's) 1) Lo-Lo pumps logic (10" setpoint for actuation) 2) Lo-Lo valves logic (10" setpoint for actuation) b. Output of A/D's goes to train A and train B 2/4 logic for "Lo-Lo valves" and "Lo-Lo pumps" EFW actuation. c. Output of 2/4 logics sets up which logic energizes. 1) If RCP's are tripped - Only EFW Actuation logic energizes - this sets up a 50% operating range setpoint for the EF-V-30's to control at and selects the operating range input to ICS with Type II compensation. 2) If RCP's are <u>not</u> tripped - EFW Actuation and setpoint logic energizes. This sets up 25" setpoint for EF-V-30's, selects the S/U range instrument as controlling input, in addition to shifting indicated operating range to type 2 compensation.	**PPT 48 OTSG "A" EF-P Auto Start & OTSG Level Selection**
3) The logic that controls EFW Actuation and RCP status relays is designed to seal in the setpoints until: a) The actuation has been cleared - AND – The EF-V-30 in question has been placed in hand to break the seal in. 4) The auto light above the controller will remain on until the seal in is broken. 5) Either the OP or SU light will be on at all times to indicate which level range is selected for EF-V-30 indication & control. d. NOTE: - "A" Train actuation starts	**ECMP 89-237** **PPT 49 EF-V-30 Setpoint Seal-in Circuits**

Examination Outline Cross-Reference Tier # 2

Evolution/System 062 AC Electrical Distribution Group # 1

K/A # K4.03 Page # 3.6-3 RO/SRO Importance Rating 2.8 3.1

Measurement Knowledge of AC distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

The plant is experiencing low system grid voltage.

Identify the ONE condition below that will cause the normal feeder breaker to the 1D 4160V Bus to trip and the emergency diesel to start and load the bus.

A. 3800V for 20 seconds.

B. 3200V for 12 seconds.

C. 2900V for 2.5 seconds.

D. 2000V for 0.5 second.

Technical Reference Technical Specification. 3.5.3.1, Engineered Safeguards Protection System Actuation Setpoints, Pages 3-37 and 3-37a, Amendments 159 and 224 respectively.

Open Exam Reference None.

Learning Objective IV.G.06.07

Question Source ☐ New ☑ Bank Question # SR4G06-07-Q02.

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the Degraded Voltage setpoint is 3760 volts.

Distracter is plausible because the time given is greater than the degraded voltage timer setpoint of 10 seconds and the voltage is very close to the maximum voltage of 3773 volts permitted by tech specs.

B CORRECT. The voltage and times given are below the setpoint voltage of 3760 volts and greater than the timer setpoint of 10 seconds.

C INCORRECT because the loss of voltage setpoint is 2400 volts.

Distracter is plausible because the given voltage is close to the maximum voltage of 2860 volts permitted by tech specs and the time is greater than the loss of voltage timer setpoint of 1.5 seconds.

D INCORRECT because the loss of voltage timer setpoint is a minimum of 1 second by tech specs.

Distracter is plausible because the voltage given is below the loss of voltage setpoint given by tech specs.

Comments None.

3.5.3 ENGINEERED SAFEGUARDS PROTECTION SYSTEM ACTUATION SETPOINTS

Applicability:

This specification applies to the engineered safeguards protection system actuation setpoints.

Objective:

To provide for automatic initiation of the engineered safeguards protection system in the event of a breach of Reactor Coolant System integrity.

Specification:

3.5.3.1 The engineered safeguards protection system actuation setpoints and permissible bypasses shall be as follows:

Initiating Signal	Function	Setpoint
High Reactor Building Pressure (1)	Reactor Building Spray	$\leq$ 30 psig
	Reactor Building Isolation	$\leq$ 30 psig
	High-Pressure Injection	$\leq$ 4 psig
	Low-Pressure Injection	$\leq$ 4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	$\leq$ 4 psig
Low Reactor Coolant System Pressure	High Pressure Injection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Low Pressure Injection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Reactor Building Isolation	$\geq$ 1600 psig(2)
4.16 kv E.S. Buses Undervoltage Relays		
Degraded Voltage	Switch to Onsite Power Source and load shedding	3760 volts (4)
Degraded voltage timer		10 sec (5)
Loss of voltage	Switch to Onsite Power Source and load shedding	2400 Volts (6)
Loss of voltage timer		1.5 sec (7)

(1) May be bypassed for reactor building leak rate test.

(2) May be bypassed below 1775 psig on decreasing pressure and is automatically reinstated before 1800 psig on increasing pressure.

(3) May be bypassed below 925 psig on decreasing pressure and is automatically reinstated before exceeding 950 psig on increasing pressure.

(4) Minimum allowed setting is 3740 v. Maximum allowed setting is 3773 v.

(5) Minimum allowed time is 8 sec. maximum allowed time is 12 sec.

(6) Minimum allowed setting is 2200 volts, maximum allowed setting is 2860 volts

(7) Minimum allowed time is 1.0 second, maximum allowed time is 2.0 seconds.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a LOCA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressures to prevent spurious initiation (Reference 1).

Low Reactor Coolant System Pressure

The basis for the 1600 and 500 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation. Bypass of HPI below 1775 psig and LPI below 925 psig, prevents ECCS actuation during normal system cooldown (References 1 and 2).

4.16 KV ES Bus Undervoltage Relays

The basis for the degraded grid voltage relay setpoint is to protect the safety related electrical equipment from loss of function in the event of a sustained degraded voltage condition on the offsite power system. The timer setting prevents spurious transfer to the onsite source for transient conditions.

The loss of voltage relay and timers detect loss of offsite power condition and initiate transfer to the onsite source with minimal time delay.

The minimum and maximum degraded voltage setpoint are "as found" readings.

References

(1) UFSAR, Table 7.1-3

(2) UFSAR, Section 14.1.2.10 - "Steam Generator Tube Failure"

Examination Outline Cross-Reference Tier # 2

Evolution/System 063 DC Electrical Distribution Group # 1

K/A # A3.01 Page # 3.6-7 RO/SRO Importance Rating 2.7 3.1

Measurement Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor startup in progress.
- Operating crew is raising power from 20%.
- FW-P-1A is operating.
- All major equipment is available.

Based on these conditions identify the ONE selection below that describes an occurrence that would indicate the existence of a problem in the "A" 125/250V DC Distribution System.

A. MAP K-3-4, MN TURB DC OIL PMP STRT/TRBL, actuates.

B. Loss of control power indication for all Reactor Coolant Pumps.

C. FW-V-7A and FW-V-7B, Feedwater Pump Recirculation Valves, both open fully.

D. RCP Seal Return Valve MU-V-26 fails open while ESAS Status Panel indicates MU-V-26 closed.

Technical Reference 1202-9A, Section 1.0.A – Bullet 7, Page 2, Rev. 44.

Open Exam Reference None.

Learning Objective IV.G.10.25

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. The Main Turbine DC Oil Pump will lose DC power resulting in the trouble alarm.

B INCORRECT because this indication loss is for the loss of B DC.

Distracter is plausible because A DC supplies control power to the 1A and 1B 7KV feeder breakers thus the feeder breakers would lose control power indications.

C INCORRECT because this is an automatic action for the loss of B DC.

Distracter is plausible because these valves utilize DC control power and will come open on a loss of B DC.

D INCORRECT because this is an automatic action for the loss of B DC.

Distracter is plausible because this valve utilizes DC control power and fails open on loss of B DC.

Comments None.

NOTE
Partial loss of DC caused by a blown fuse to one of the DC Distribution Panels may give similar symptoms. This procedure does **NOT** provide specific guidance for a blown fuse.

1.0 **SYMPTOMS**

A. Loss of Main Distribution Panel 1A as indicated by alarms:

- AA-3-2, 7KV Bus Trouble
- AA-3-3, 4KV BOP Bus Trouble
- AA-3-5, 480V BOP Bus Trouble
- A-1-7, Battery 1A Discharging (Rate above 100 amps)
- A-2-7, Battery Charger 1A/1C/1E Trouble
- A-3-7, Inverter 1A/1C/1E Trouble
- K-3-4, MN Turb. PC oil pmp strt/troub
- L-1-3, Voltage Regulator DC Loss
- B-3-1, 4KV ES Bus Trouble
- NN-3-1, 230KV Substation Trouble (loss of DCA)
- PRF-1-1-1, CRDM Breaker Test Trouble (loss of shunt trip)
- H & V, A-4-2 Cont. Bldg. Batt. Chargers A Damper Tbl, Fire-Smoke
- Loss of breaker status lights at control switches

B. Loss of Main Distribution Panel 1A will result in the following:

- Loss of all power on the "A" Distribution System.
- Inability to remotely trip or close breakers on A ESAS System.
- Loss of Engineered Safeguards Distribution Panel 1E.
- Loss of ES Diesel Generator Dist. Pnl. 1P.
- Loss of 230KV Substation Dist. Pnl. DCA.
- Loss of Distribution Panel 1C.

Form ES-401-5 **Written Exam Question Worksheet** Q # 023

Examination Outline Cross-Reference Tier # 2

Evolution/System 063 DC Electrical Distribution Group # 1

K/A # A1.01 Page # 3.6-7 RO/SRO Importance Rating 2.5 3.3

Measurement 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.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor is tripped.
- Loss of off-site power (LOOP).
- EG-Y-1A and EG-Y-1B failed to start.
- Station Battery 1A load shedding has just been completed in accordance with OP-TM-AOP-020, Loss of Station Power.
  - Load was reduced from 300 Amps down to 150 Amps.

If battery load was maintained at 300 Amps, the battery would be discharged in 3.5 HOURS from now.

Based on these conditions identify the ONE selection below that describes how long Battery 1A will be able to supply the REMAINING loads.

A. 3.5 hours.

B. More than 3.5 hours, but less than 7 hours.

C. 7 hours.

D. More than 7 hours.

Technical Reference MAP A-1-7, Battery 1A Discharging, Page 2, Rev. 15.
11.2.01.017, Vital AC/DC Distribution System, PPT 12, Rev. 16.

Open Exam Reference None.

Learning Objective IV.G.10.12

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # Harris 2004 Q-054

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because with the reduced battery discharge rate, remaining time for the battery will be beyond 3.5 hrs.

Distracter is plausible because the original battery life was given as 3.5 hours and if the reduced load is not taken into account then the battery life would remain at 3.5 hours.

B INCORRECT because with the reduced battery discharge rate, remaining time for the battery will be 9 hrs.

Distracter is plausible because if the non-linear relationship between discharge rate and battery life where the extension of life is less than the reduction of discharge rate is assumed, then the life would fall some where between 3.5 and 7 hours.

C INCORRECT because with the reduced battery discharge rate, remaining time for the battery will be 9 hrs.

Distracter is plausible because with the discharge rate cut in half, if a linear relationship between change in discharge rate and change in battery life is assumed, then the battery life would be 7 hours.

D CORRECT MAP A-1-7 gives the expected battery life for a 150 amp discharge rate as 9 hours.

Comments Modified Bank -Harris 2004 NRC RO Exam Q-054.

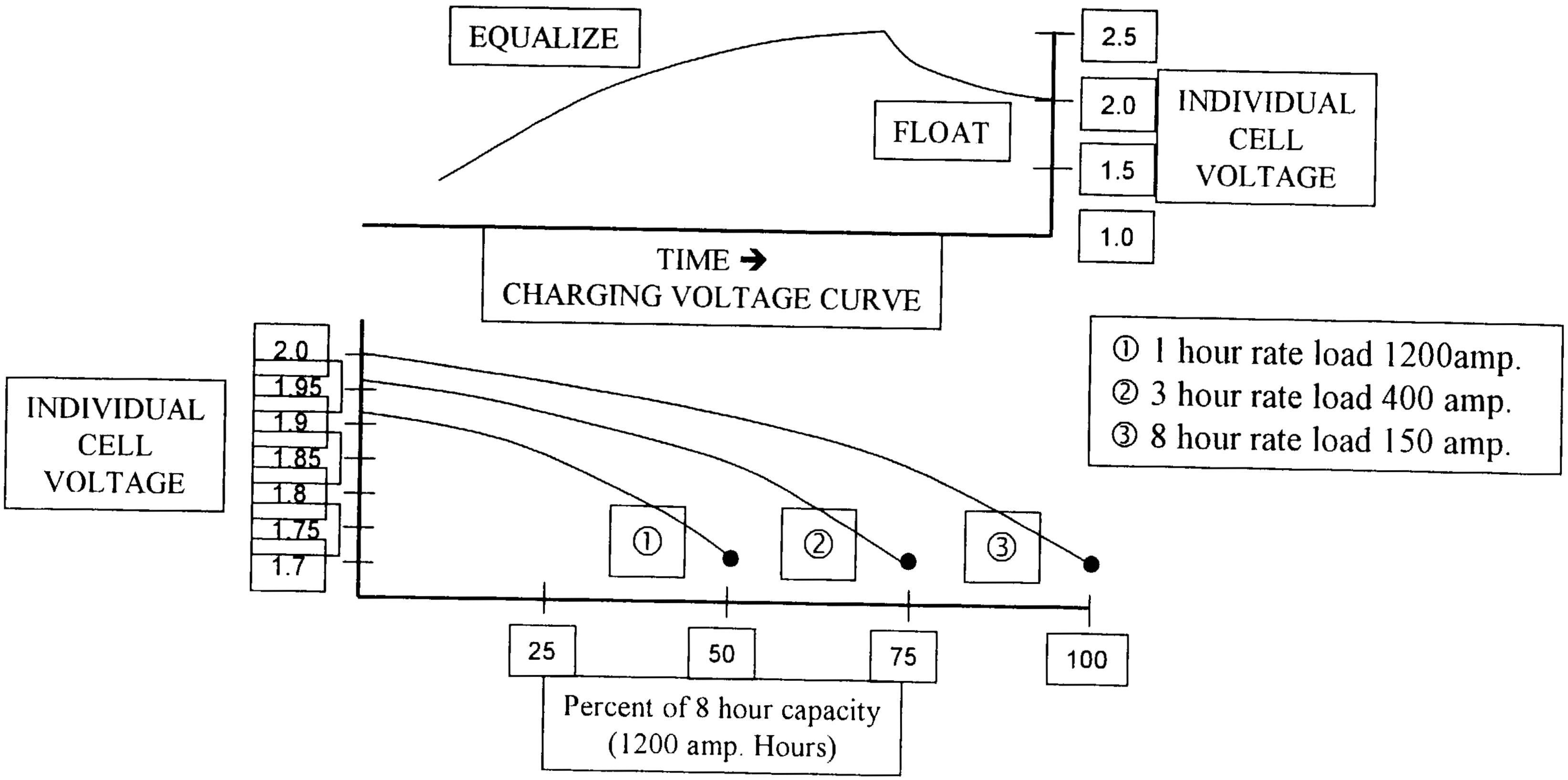
MANUAL ACTION REQUIRED:

1. Determine status of the 1A battery.
   1.1 Check the ammeters on the Main DC distribution panel A

Amps	Total time to bank depletion
150	9 hours
200	6 hours
250	4 1/2 hours
300	3 1/2 hours
350	2 1/2 hours

   1.2 Check the voltmeters at the 1A battery ground detector.
   1.3 If voltage is available on the 1P 480V bus, restore the 1A, 1C and 1E inverters to their AC supply per MAP A-2-6.
   1.4 Place the spare charger in service on the side with the highest discharge rate.
2. If a battery is being significantly discharged.
   2.1 Extend the life of the battery bank by removing unnecessary DC loads from the battery. Refer to OP-TM-AOP-020, Attachment 3.
   2.2 If battery voltage drops less than 105V,
   - go to 1202-9A - Loss of "A" DC distribution.
   - follow up actions per Tech Spec 4.6.2.a requires performance of SP 1301-5.8, "Station Battery Quarterly"
3. Check Panel CT-5 energized and breaker 10 closed. This is power to the 1A/1B battery ground detector.
4. **If** battery voltage < 125VDC **or** current > 100 amps for greater than 5 minutes, **then CONTACT** the EDM **and DETERMINE** battery operability IAW LS-AA-105 within 8 hours.
5. **If** less than two battery charges are in service, **then ENTER** the action statement for Tech Spec. 3.7.

Battery Capacity / Recharge



Form ES-401-5 **Written Exam Question Worksheet** Q # 024

Examination Outline Cross-Reference Tier # 2

Evolution/System 064 Emergency Diesel Generator (ED/G) System Group # 1

K/A # K6.07 Page # 3.6-9 RO/SRO Importance Rating 2.7 2.9

Measurement Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- NO Emergency Diesel Generators operating.

Event 1:
- Alarm A-1-2, DIESEL GEN 1A TROUBLE, actuation.

Field report:
- STARTING AIR PRESSURE LOW alarm is actuated on EG-Y-1A local panel.
- Air compressor has been isolated from BOTH starting air receivers in order to terminate the leak.
- Starting air pressure is now steady at 120 psig.

Event 2:
- Loss of offsite power (LOOP) concurrent with ES actuation.

Based on these conditions identify the ONE selection below that completes the following statement:
Emergency Diesel Generator EG-Y-1A will ____________________.

A. start and meet all design basis requirements.

B. start but NOT reach full speed to pick up electrical load within 10-second requirement.

C. attempt to start but trip when the Start Failure Relay (SFR) actuates.

D. NOT attempt to start because it is locked out by the Shutdown Relay (SDR).

Technical Reference 1107-3, Diesel Generator, Section 2.1.5 Air System - Limits/Precautions, Page 16, Rev. 110.

Open Exam Reference None.

Learning Objective IV.G.8.19

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. Based on this low air pressure, EDG cranking RPMs are lower and it takes longer to startand accelerate to full speed. This lengthens how long it takes to begin electrical loading. The diesel generator must be capable of starting and loading within 10-seconds.

Distracter is plausible based on misconception that there is no impact on diesel starting until starting air pressure is less than 100 psig.

B CORRECT. The EDG is not operable with starting air pressure less than 175 psig. The diesel is capable of starting with air pressure as low as 100 psig, but depending how low air pressure is, may not be capable of

5. Make-up to the expansion tank can occur during engine operation. Level Indication will not be available; therefore, the operator shall estimate the level based on visual observations through the tank vent. The vent is large enough to fill and observe the level.

6. In the event of a large leak and engine operation is required, the following sources of make-up should be used/considered: (Freeze protection is not required while engine is running)

   a. Mixed ethylene glycol.

   b. Demineralized Water.

   c. Any clean water source.

2.1.5 Air System - Limits/Precautions

a. If diesel fails to start during an ES condition, the diesel will continue to crank until the starting air is consumed. To stop the diesel from cranking, place the Control Room STARTING switch to the MANUAL (EXERCISE) position. If diesel is stopped, it is necessary to wait 60 seconds before attempting restart.

NOTE

The diesel generator has the ability to start and load with an air pressure as low as 100#. Based on the physical condition of the Air Start System, the affected diesel generator can be considered in reduced availability and may not meet 10 second start/load criteria. Contact system engineering to address operability under degraded conditions when below 175#.

b. Alarm (DGA/B-3-1) for starting air pressure low provides indication that a problem exists in the starting air system. If receiver air pressure drops below 175#, and EG-Y-1A(B) is not running, then declare EG-Y-1A(B) inoperable per Tech Spec requirements.

2.1.6 EDG Room Ventilation

a. If the EDG room ventilation is out of service, Then Declare the associated EDG inoperable and enter the appropriate Tech Spec Clock per section 3.7.2.

b. Operation of an EDG without room ventilation requires action within 1 hour, notification to security and actions to open the respective doors per procedure 1104-45P.

c. The gravity damper is only critical for EDG operation upon a loss of room ventilation. The damper is to be positioned open to assist with heat removal.

meeting the 10-second start/load criteria. Final air pressure in the stem conditions was selected to be 120 psig to ensure the diesel could still start - but would take longer.

C INCORRECT because the start failure relay will not actuate.

Distracter is plausible because if the diesel did not start, then the start failure relay would actuate.

D INCORRECT. SDR does not actuate on any problem related to starting air pressure. The SDR will trip on a start failure, but will not prevent diesel start attempt on low air presssure.

Distracter is plausible because the SDR will shutdown the diesel and prevent it from restarting on a start failure.

Comments None.

Form ES-401-5 **Written Exam Question Worksheet** Q # 025

Examination Outline Cross-Reference Tier # 2

Evolution/System 073 Process Radiation Monitoring (PRM) System Group # 1

K/A # K3.01 Page # 3.7-15 RO/SRO Importance Rating 3.6 4.2

Measurement Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ✔ PRA Related **Correct Answer** C.

Plant conditions:
- WDG-T-1B Waste Gas Tank radioactive gas release in progress.

Event:
- Fuse failure de-energizes RM-A-7, Waste Gas Release Monitor.

Based on these conditions identify the ONE selection below that describes:
(1) Effect of this malfunction on the radioactive effluent release.
(2) Design feature that produces/enables the effect.

A. (1) Gas release will continue.
(2) RM-A-7 interlocks are "energize to actuate."

B. (1) Gas release will continue.
(2) RM-A-7 interlock circuit is still energized from separate external power supply.

C. (1) Gas release will NOT continue
(2) RM-A-7 monitor is de-energized with its interlock defeat switch in NORMAL position.

D. (1) Gas release will NOT continue.
(2) RM-A-7 and WDG-V-47 solenoid have common power supply.

Technical Reference 209-707, Radiation Monitoring Interlocks, Rev. 7.
209-324, WDG-V-47, Rev. 7.

Open Exam Reference None.

Learning Objective IV.E.06.10

Question Source ✔ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the WDG-V-47 closes when the monitor is de-energized.

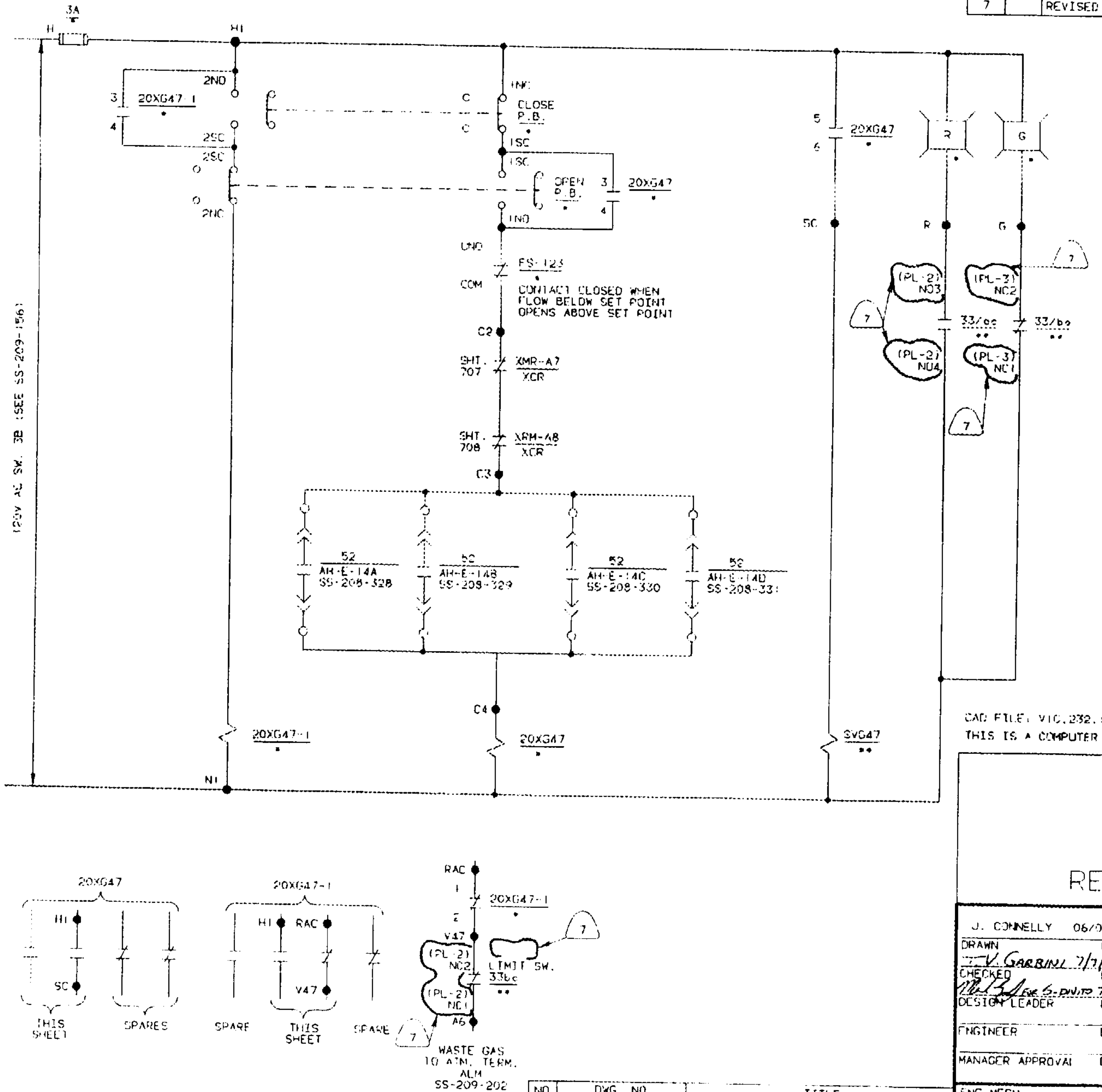
B INCORRECT because the WDG-V-47 closes when the monitor is de-energized.

C CORRECT answer.

D INCORRECT because the power supplies are not common

Comments Similar to Audit Exam Q-080 - addresses liquid monitor RM-L-7 de-energized with interlocks defeated.

REV.	ZONE	DESCRIPTION
7		REVISED TO INCORPORATE FCN C108595



NOTES

1. * - DEVICES MOUNTED ON RW PANEL-(AUX. BLDG.) INCLUDING AUX. RELAYS.
   ** - DEVICES MOUNTED LOCALLY (NEAR OR AT MOTOR, VALVE, ETC.)

REFERENCE DWGS.

INDEX & LEGEND SS-209-151 THRU 155
LOGIC DIAGRAM S-203-F254
FLOW DIAGRAM C-302-694

CAD FILE: VIC.232.;7,1002,001-,070)
THIS IS A COMPUTER GENERATED DWG. DO NOT REVISE IT MANUALLY

REGULATORY REQUIRED

NO.	DWG. NO.	TITLE
REFERENCES		

J. CONNELLY 06/05/94
DRAWN DATE
V. Garbini 7/7/94
CHECKED DATE
DESIGN LEADER DATE 7-1-94
ENGINEER DATE
MANAGER APPROVAL DATE
ENG. MECH.

GPU Nuclear/GAI

WASTE HANDLING SYSTEM
ELECTRICAL ELEMENTARY DIAG.
WASTE GAS RELEASE VALVE WDG-V-47

TMI UNIT 1
DWG. NO. 209-324
SH. .
REV. 7
SCALE: NONE
BA .
TASK .

DWG. NO. 209-324
SH.

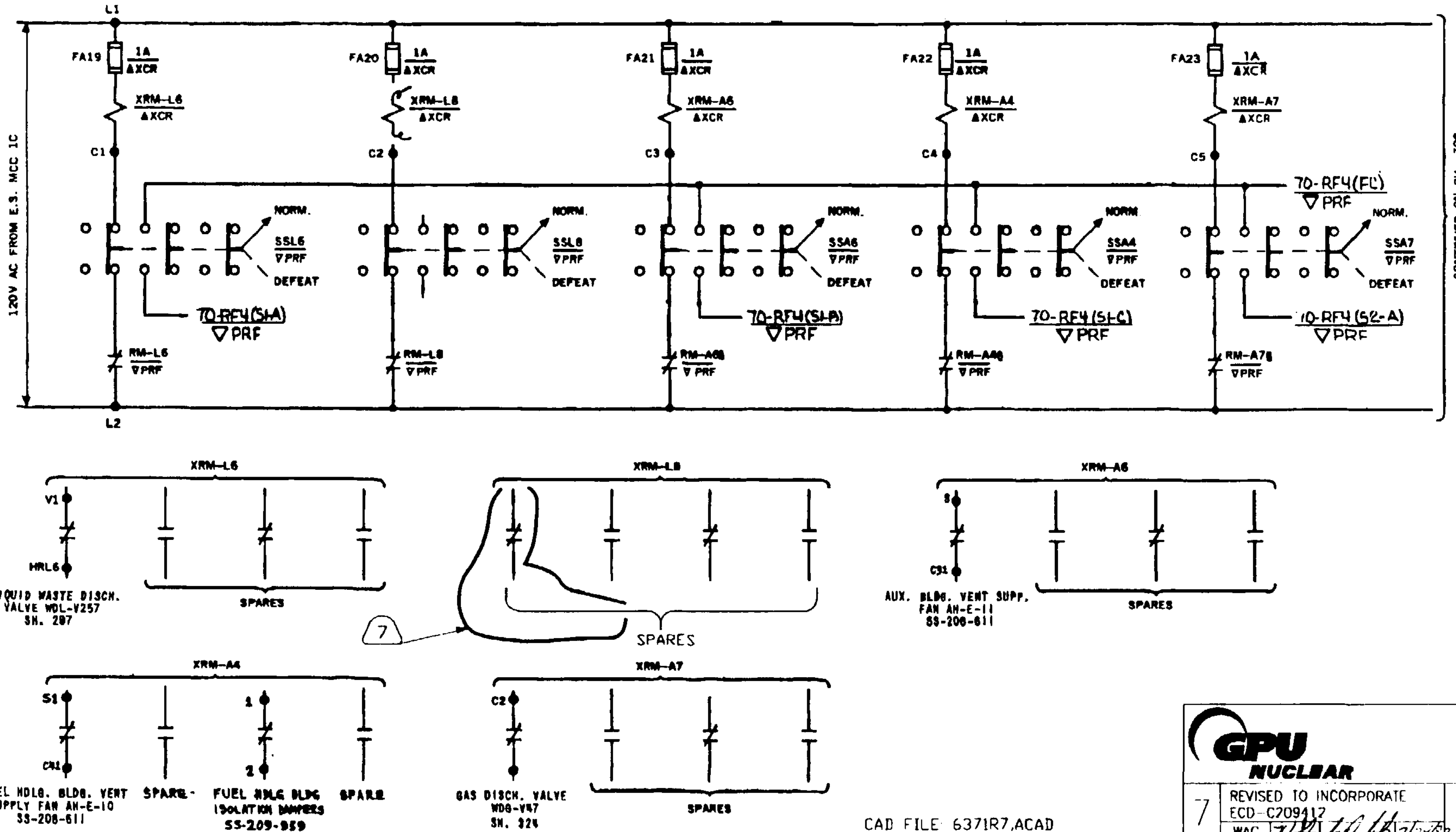
REFERENCE DWGS.:
LEGEND: SS-208-001
SW. DEVELOPMENT: SH. 009
INDEX: SS-209-001

NOTES:
1. ALL XRM RELAYS ARE CLARK 8U4-2
2. RM-RELAY CLOSE ON HIGH RADIATION OR LOSS OF POWER.

METROPOLITAN EDISON COMPANY THREE MILE ISLAND NUCLEAR STATION UNIT 1	MADE RF CHK'D. FFB SR. CF. GC CF. DPN JCG	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.	
ELECTRICAL ELEMENTARY DIAGRAM		4192 (WORK ORDER)	SS-209-707 (SIZE DRAWING) / 7 (REV.)
D.C. & MISCELLANEOUS			

REGULATORY REQUIRED

RADIATION MONITORING INTERLOCKS



CAD FILE 6371R7.ACAD
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GPU NUCLEAR

REV	DRAFT	CHECK	APPROVED	DATE	APP
7	REVISED TO INCORPORATE ECD-C209412 WAC				

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Form ES-401-5 **Written Exam Question Worksheet** Q # 026

Examination Outline Cross-Reference Tier # 2

Evolution/System 076 Service Water System (SWS) Group # 1

K/A # A2.01 Page # 3.4-50 RO/SRO Importance Rating 3.5 3.7

Measurement 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: Loss of SWS.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ✔ 55.41 .5 55.43

Proposed Question ✔ RO SRO ✔ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- Drought and heat wave has resulted in slowly reduced river water level and rising river water temperature.
- Operating crews have been in OP-TM-AOP-005, River Water Systems Failures, for several days.

Event:
- Auxiliary Operator report:
  - ISPH Pump Bay Water Level is 281 feet elevation.
  - River Water temperature is 95 degrees F.

Based on these conditions identify the ONE selection that describes required action(s), and the problem.

A. Initiate plant shutdown to be at CSD within the next 36 hours.
Margin of safety for ECCS heat removal capability is being challenged.

B. Trip the reactor, then stop all four Reactor Coolant Pumps.
RCP failure may occur due to inadequate cooling of components.

C. Initiate actions to provide a redundant Fire Service Water supply.
Ability to maintain adequate fire header pressure is in jeopardy.

D. Reduce the number of running river water pumps to the minimum requirement.
NPSH requirements may not be met with all river water pumps running.

Technical Reference OP-TM-AOP-005, River Water Systems Failures, Page 1, Rev. 3.
SDBD-T1-533/543, System Design Basis Document for DRWS and DCCS, Page 3-20, Rev. 2.

Open Exam Reference None.

Learning Objective IV.B.02.01

Question Source ✔ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. Refer to AOP-005 Step 3.4 actions. Maximum river water temperature is based on ECCS heat removal capabilities.

B INCORRECT because the requirement is to execute a controlled shutdown to minimize risk of transient conditions that could lead to ECCS operation.

Distracter is plausible because this is an action in loss of all NR and SR Pumps section.

C INCORRECT because the requirement is to execute a controlled shutdown to minimize risk of transient conditions that could lead to ECCS operation.

Distracter is plausible because this is an action in loss of all NR and SR Pumps section.

D INCORRECT because the requirement is to execute a controlled shutdown to minimize risk of transient conditions that could lead to ECCS operation.

Distracter is plausible because this is an action in low river water level section.

Comments None.

Exelon Nuclear

RIVER WATER SYSTEMS FAILURES

1.0 ENTRY CONDITIONS

1.1 ISPH pump bay level < 277' (SR-LI-1172) **or**

1.2 Failure of York Haven Dam **or**

1.3 River water temperature > 90°F **or**

1.4 Failure of all secondary and nuclear services river water pumps.

2.0 IMMEDIATE ACTIONS

None

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE: SR-LI-1172 is used to measure river water pump bay level above 271' elevation.	
☐ 3.1 **IAAT** all NR and SR pumps are inoperable **or** ISPH pump bay water level < 271', **then** ____ **TRIP** the reactor. ____ **INITIATE** OP-TM-EOP-001 Reactor Trip. ____ **GO TO** Section 4.3.	
☐ 3.2 **IAAT** ISPH pump bay water level < 276', **then INITIATE** a plant shutdown IAW 1102-4 "Power Operations" to be at HSD IAW TS 3.0.1 requirements.	
____ 3.3 **VERIFY** ISPH pump bay water level > 277'.	____ **GO TO** Step 4.2.
NOTE: If available use local measurement of river water temperature per Attachment 4.	
☐ 3.4 **IAAT** river water temperature ≥ 95°F, **then INITIATE** a plant shutdown IAW 1102-4 "Power Operations" to be at CSD within 36 hours.	
____ 3.5 **VERIFY** river water temperature (A0089) < 90°F.	____ **GO TO** Step 4.1.
END	

Basis: The original cooling water design temperature of 95° F was the basis for B&W 's design of the DH cooler. This DC System temperature value assumed a worst case cooling in the Decay Heat Service Cooler with a Decay Heat River System tube side river inlet temperature of 85° F. These numbers were based on a worst-case design base condition for the heat exchanger (46% clean). (B079)

On several occasions, the river temperature (DR) exceeded this 85° F value which in turn elevated the DC temperature. Therefore a 95° F maximum river water (DR) temperature was evaluated. [See Table 3 below] (A058, p2)[A008]

Basis for this maximum temperature included TMI-2 Tech Specs, maximum recorded temperature at TMI 91.3° F and maximum recorded Susquehanna River governmental water quality records of 93.2° F. Minimum river water temperature is 33° F. (A002; F012; B071; A057; B054; A066)

Accident analyses were performed with river water temperatures of 95° F. Table 3 below identifies the results of these analyses.

Table 3
Impact of 95° F DR Flows on DC Temperature

DR Flow GPM (Tube Side)	DR Inlet Temperature ° F	DR Outlet Temperature ° F	DCCW Outlet Temperature ° F	DCCW Inlet Temperature ° F	Reference
6,000	95	119.71	101.0	150.22	A380, A395
7,500	95	123.4	99.5	164.9	A058, A087
8500	95	120.5	98.9	161.9	A058, A087

In both conditions, at both the design river flow of 7500[4] gpm and at an increased flow of 8500 GPM, the DCCW water temperature is less than 100° F which is compatible with the requirements of the components to be cooled and which does not affect accident analysis.

[4] DR flow is now 6500 gpm (A387 and A394) for the new Johnston DR Pumps that were installed in 13R.

Form ES-401-5 **Written Exam Question Worksheet** Q # 027

Examination Outline Cross-Reference Tier # 2

Evolution/System 078 Instrument Air System (IAS) Group # 1

K/A # K1.04 Page # 3.8-19 RO/SRO Importance Rating 2.6 2.9

Measurement Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor.

10 CFR Part 55 Content ☑ 55.41 .2 to .9 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Identify the ONE condition below that describes when the backup cooling water supply (Fire Service Water) is used to cool Instrument Air compressors IA-P-1A and IA-P-1B.

A. Low SCCW System pressure.

B. High SCCW System temperature.

C. Tripping of all 3 SCCW pump breakers.

D. High Instrument Air compressor temperature.

Technical Reference Lesson Plan 11.2.01.053, Instrument and Control Air, Pages 5 and 6, Rev. 21.

Open Exam Reference None.

Learning Objective IV.D14.07

Question Source ☐ New ☑ Bank Question # AO4D14-07-Q03

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because low SCCW system pressure will not cause SC-V-57A/B and SC-V-58A/B, fire service water to IA-P-1A/B, to open.

Distracter is plausible because low SCCW system pressure would result in IA-P-1A/B receiving less cooling flow.

B INCORRECT because high SCCW system temperature will not cause SC-V-57A/B and SC-V-58A/B, fire service water to IA-P-1A/B, to open.

Distracter is plausible because high SCCW system temperature would result in IA-P-1A/B cooling water temperature rising.

C CORRECT. All three SCCW pump breakers open will open SC-V-57A/B and SC-V-58A/B, fire service water to IA-P-1A/B.

D INCORRECT because instrument air compressor temperatures are not interlocked with SC-V-57A/B and SC-V-58A/B.

Distracter is plausible because high instrument air compressor temperature would indicate a need for supplemental cooling.

Comments None.

Content/Skills	Activities/Notes
3. System Description – Standby Instrument Air	
a. Two 100% capacity compressors.	
1) 250 SCFM for each compressor.	
2) Motor driven by V-belts.	
a) IA-P-1A powered from 1A 480V ES MCC	
b) IA-P-1B powered from 1B 480V ES MCC	
3) Non-Lubricated	
a) Teflon Piston Rings	
b) Graphite Wear Runners	
4) Dual Control	
a) Hand or Auto Locally	
b) Manual from the Control Room	
c) Local control will be in auto (Standby)	
d) IA-P-1B can be controlled remotely from breaker.	
5) Will load and unload @ 95-110#	
a) Setpoints raised from 85-95 to 95-110# as a result of CR 203857.	
b) No longer need to adjust IA-P-1A/B set points when IA-P-4 taken out of service.	
6) Water-cooled, never operate unless cooling water is established.	
a) SSCC normally	

b) Fire Service in an emergency, loss of all 3 SSCC pumps from breaker Contacts.

c) Do not add oil when compressor is running.

7) Located in IB basement.

b. Intake Filter/Silencer

(1) Dry wool felt element.

c. After-cooler (2)

1) Removable tubes

2) 9 GPM cooling water flow

a) SSCC normally

b) Fire Service is the backup

d. Moisture Separator

1) Baffled plates

2) Gage glass and drain trap

a) Blown down once a shift (manually)

e. Air Receivers (2) (IA-T-1A/B)

1) Vinyl coated interior

2) Pressure gage

3) Relief Valve – opens at 120#

4) Drain trap and Receiver Blowdown valve IA-V-10A/B

The previously existing Armstrong Inverted Bucket Traps have been replaced with liquid-level float traps. A "Y" strainer has also been added upstream of the trap to prevent the trap from being clogged with rust/dirt.

a) Blowdown Receiver (IA-V-10A/B) and After cooler Trap once a shift per secondary AO logs.

Form ES-401-5 **Written Exam Question Worksheet** Q # 028

Examination Outline Cross-Reference Tier # 2

Evolution/System 103 Containment System Group # 1

K/A # A4.04 Page # 3.5-19 RO/SRO Importance Rating 3.5 3.5

Measurement Ability to manually operate and/or monitor in the control room: Phase A and phase B resets.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor tripped.
- RCS LOCA inside the RB caused RB pressure to rise to 35 psig.
- All ES actuations and systems operated as designed.
- Train A and Train B 30 psig RB Isolation has actuated.
- Current RB pressure is 15 psig.

Based on these conditions, identify the ONE selection below that describes MINIMUM required actions to enable re-opening NS-V-15, NS cooling to the Reactor Building.

A. Bypass 1600 psig and 500 psig ES actuations.

B. Bypass 1600 psig and 500 psig ES actuations.
Defeat 4 psig RB Isolations.

C. Defeat 4 psig ES Actuations.
Reset 30 PSIG RB Isolation.

D. Reset 30 PSIG RB Isolation.

Technical Reference Lesson Plan 11.2.01.084, Nuclear Services Closed Cooling System, Page 12, Rev. 10.

Open Exam Reference None.

Learning Objective IV.E.24.16

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because NS-V-15 is closed by 30# actuation, not 1600# or 500# actuation.

Distracter is plausible because 1600# and 500# actuations have to be bypassed in order to gain control of other ES components.

B INCORRECT because NS-V-15 is closed by 30# actuation, not 1600#, 500# or 4# actuations.

Distracter is plausible because 1600#. 500# and 4# actuations have to be bypassed in order to gain control of other ES components.

C INCORRECT because 4# actuation does not close NS-V-15.

Distracter is plausible because 30# actuation does close NS-V-15 and will have to be reset to regain control of NS-V-15.

D CORRECT. Resetting 30# actuation will allow control to be restored to NS-V-15.

Comments Question addresses manual control room operation required to re-establish control of a component interlocked to close on 30 psig RB Pressure (equivalent to Westinghouse Phase B) isolation system. This is required knowledge for the Examinee to be able to manually operate the equipment himself, or to monitor another operator performing this function.

Content/Skills	Activities/Notes

b) Ex: NS-P-1A and 1C pumps start on ES, load block 3, and 1B pump does not restart when power is restored.

3) Line Break/Isolation

a) Surge tank low level alarm received during an ES condition activates the Line Break Isolation circuit, resulting in closure of NS-V-4, NS-V-15 and NS-V-35.

b) This action isolates cooling water flow to the Reactor Building preventing the Reactor Building atmosphere from escaping through a ruptured cooling water line inside the Reactor Building.

c) Line Break Isolation circuit (L.B.I) must be manually reset by the operator when an ES condition is cleared and the surge tank level returns to its minimum normal level.

d) This allows NS-V-4, NS-V-15, and NS-V-35 to be reopened by the operator.

e) L.B.I. broken into two (2) channels. Channel "A" closes NS-V-4 and NS-V-15, and Channel "B" closes NS-V-35 and NS-V-15.

f) Insures the inlet valve (NS-V-15) and at least one of the outlet valves (NS-V-4 or NS-V-35) close.

4) 30# ES Containment Isolation

a) NS-V-4, 15, and 35 close

b) NS-V-4 Train-A, NS-V-15 Train-A/B, and NS-V-35 Train-B

7. Interfacing Systems

a. Reactor Coolant Pumps

1) The RCP motor is totally enclosed and air cooled. Hot air from the stator is circulated through 2 air-to-water heat exchangers which give up heat to the NS system.

Form ES-401-5 **Written Exam Question Worksheet** Q # 029

Examination Outline Cross-Reference Tier # 2

Evolution/System 001 Control Rod Drive System Group # 2

K/A # K2.02 Page # 3.1-3 RO/SRO Importance Rating 3.6 3.7

Measurement Knowledge of bus power supplies to the following: One-line diagram of power supply to trip breakers.

10 CFR Part 55 Content ✓ 55.41 .2 to .9 ☐ 55.43

Proposed Question ✓ RO ☐ SRO ✓ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- RPS surveillance in progress on C RPS channel.
- Breakers/electronic trips associated with C RPS channel were tripped.
- NLO has manually re-closed the DC breakers associated with C RPS cabinet.
- Fault Reset pushbutton has NOT been depressed on the Diamond Rod Control Panel.

Based on these conditions identify the ONE selection below that describes the IMMEDIATE effect if 120V AC Vital Bus 1D is tripped (de-energized).

A. CRD Groups 1 and 3 (ONLY) would drop.

B. CRD Groups 1, 2, 3, and 4 (ONLY) would drop.

C. CRD Groups 5, 6, and 7 (ONLY) would drop.

D. CRD Groups 1 through 7 would drop.

Technical Reference Lesson Plan 11.2.01.013, Control Rod Drive, PPT 22, Rev. 7.

Open Exam Reference None.

Learning Objective IV.E.13.07

Question Source ☐ New ✓ Bank Question # NRC Exam #20 Q-044.

☐ Modified Bank Parent Question #

Question NRC Exam History TMI 2001 Q-044

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because Groups 1 and 3 still are supplied power via the C RPS cabinet DC breakers.

Distracter is plausible because vital bus D does supply half of the power to the safety groups.

B INCORRECT because Groups 1 through 4 are still supplied power via C RPS cabinet DC Breakers.

Distracter is plausible because vital bus 'D' does supply half of the power to the safety groups.

C CORRECT. The power to the regulating groups is lost due to not resetting C RPS electronic trips and loss of power to D RPS electronic trips.

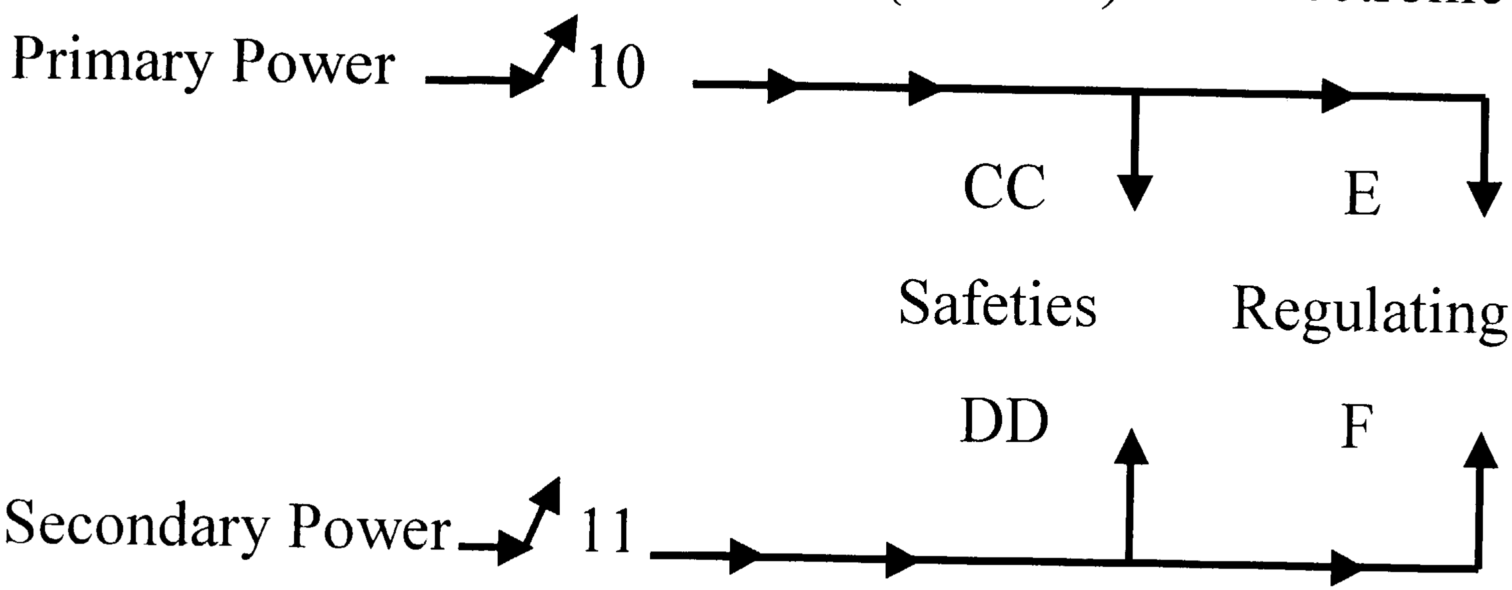
D INCORRECT because groups 1 through 4 are still supplied power via the C RPS cabinet DC Breakers.

Distracter is plausible all secondary power is lost to all CRD groups with the C electronic trips not reset.

Comments 2001 TMI NRC Exam Q-044.

RPS CHANNELS AND ASSOCIATED CRD BREAKERS

RPS Channel	CRD Breaker
A	10
B	11
C	CC (CB1&2) & E electronic trip
D	DD (CB3&4) & F electronic trip



Examination Outline Cross-Reference Tier # 2

Evolution/System 015 Nuclear Instrumentation Group # 2

K/A # K6.01 Page # 3.7-7 RO/SRO Importance Rating 2.9 3.2

Measurement Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Sensors, detectors, and indicators.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor startup in progress.
- Reactor is supercritical, power level entering the Intermediate Range.
- Operator is manually withdrawing control rods to achieve +1.4 DPM startup rate.

NI-11	NI-12	NI-3	NI-4
1E5 CPS	1E5 CPS	1E-10 Amps	1E-10 Amps

- All four Source and Intermediate Range NI channels indicate transient +1.0 DPM, slowly rising.
- Group 7 rods are 30% withdrawn.
- Group 8 APSRs are 30% withdrawn.

Event:
- NI-3 Rate of Change Amplifier fails HIGH.

Identify the ONE statement below that descibes the automatic response to these conditions addressing ability to withdraw AND insert rods.

A. (1) Operator can continue to withdraw control rods.
(2) Operator can insert control rods.

B. (1) Operator CANNOT withdraw control rods.
(2) Operator can insert control rods.

C. (1) Operator CANNOT withdraw control rods.
(2) Operator CANNOT insert control rods, but this does not prevent dropping the rods on a reactor trip.

D. (1) Operator CANNOT withdraw GROUP 8 APSRs.
(2) Operator CANNOT insert GROUP 8 APSRs.

Technical Reference Lesson Plan 11.2.01.082, Nuclear Instrumentation System, PPT 26, Rev. 9.

Open Exam Reference None.

Learning Objective IV.E.13.22

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the NI-3 Rate of Change Amplifier failing high will trip the +3 DPM outmotion inhibit for control rods.

Distracter is plausible because if there is a misconception that both channels of IR SUR must trip to insert the outmotion inhibit for control rods, then the assumption would be that there is no effect on rod control.

B CORRECT. The Rate of Change Amplifier failing high will trip the outmotion inhibit for rods on + 3 DPM

SUR. Insertion of control rods is not affected.

C INCORRECT because only outmotion of control rods is affected by the Rate of Change Amplifier failing high.

Distracter is plausible based on misconception that the high startup rate interlock prevents all rod motion.

D INCORRECT since only outmotion of control rods is affected by the Rate of Change amplifier failing high.

Distracter is plausible because insertion of the APSRs would result in a positive reactivity addition therefore a APSR insertion would be blocked in addition to the outmotion inhibit interlock

Comments Question addresses knowledge of the effect of a malfunction in a High SUR amplifier, and its impact on the NI system, which includes the interlock logic to initiate control rod withdrawal blocks to the CRD system. This is in concert with the connection to 10CFR55.41(7) (Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features).

Intermediate Range NI Channel Interlock Function

- Interlock provided by *either* channel (NI-3 or 4)
  - CRD out-motion inhibit (and MAP alarm) on high SUR
    - bistable actuates at 3.0 DPM increasing
    - bistable resets at 0.5 DPM decreasing
    - interlock bypassed if >10% on NI-5 or 6 and 7 or 8
      - bypass resets at <3% power

Examination Outline Cross-Reference Tier # 2

Evolution/System 016 Non-Nuclear Instrumentation Group # 2

K/A # 2.4.31 Page # 2-14 RO/SRO Importance Rating 3.3 3.4

Measurement Emergency Procedures / Plan Knowledge of annunciators alarms and indications, and use of the response instructions.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ✔ 55.41 .10 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Initial plant conditions:
- Reactor is operating at 100% power with ICS in full automatic.

Event:
- ICS signal input for RCS Loop A flow is slowly failing low.
- ICS signal input for total RCS Flow is normal.
- MAP H-3-2, SASS Mismatch, actuates, but there is NO SASS actuation.
- MAP H-2-4, Rx Inlet Delta-TC Hi, actuates.
- MAP H-1-2, OTSG A BTU Limit, actuates.

Based on these conditions, identify the ONE set of ICS Hand/Auto Stations that is required to be transferred to HAND to mitigate the consequences of this event.

A. Loop A Feedwater Master, and
Loop B Feedwater Master.

B. Loop A and Loop B Startup Feedwater valves FW-V-16A and FW-V-16B, and
Loop A and Loop B Main Feedwater valves FW-V-17A and FW-V-17B.

C. Feedwater Pump FW-P-1A (ONLY), and
Loop A Feedwater Valves FW-V-16A and FW-V17A ONLY.

D. Both Feedwater Pumps FW-P-1A and FW-P-1B, and
Loop A Feedwater Valves FW-V-16A and FW-V-17A.

Technical Reference OP-TM-MAP H0204, Rx Inlet DTC Hi, Page 1, Rev. 0.
D553731, ICS Feedwater Subsystem Analog Logic Drawing, Rev. Q.

Open Exam Reference None.

Learning Objective IV.E.27.60, IV.E.27.52.c.

Question Source ☐ New ✔ Bank **Question #** 2003 TMI SRO Audit Q-078

☐ Modified Bank **Parent Question #**

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the faulted signal comes in below the Feedwater Loop Masters, as evidenced by actuation of OTSG A BTU Limit MAP.

Distracter is plausible because one possible response to MAP H-2-4, Delta Tc High, is to place both FW Loop Masters in hand.

B INCORRECT because the faulted signal goes to both the feedwater valves and the feedwater pumps.

Distracter is plausible because one possible response to MAP H-2-4, Delta Tc High, is to place the feedwater valves in manual.

C INCORRECT because the faulty signal is being fed to both the A and B feed pumps and feedwater valves.

Distracter is plausible because the fault is from RCS Loop A flow signal.

D CORRECT. This signal failure affects one ICS FW Train plus both FWP control signals - requiring 4 hand/auto stations to be placed in manual to totally block use of the failed signal (Both FW Pumps plus both Loop A FW Valves).

Comments Added dTC alarm to 2003 SRO Audit Q-078 to better fit this KA.
Added BTU Limit alarm to ensure distracter A is not correct.

1.0 SETPOINTS

– ΔTc ±5°F
from selected Loop A/B NR Tc at RC-5A/B MS2

2.0 CAUSES

– Unequal RCS flow between loops (Ex: RCP trip)

– Unequal F.W. flow to the Steam Generators

3.0 AUTOMATIC ACTIONS

– For RCP trip, ΔTc Error Signal is boosted
to re-ratio feedwater → *lowers feedwater flow in the
affected loop.

(*relative to unaffected loop)

4.0 MANUAL ACTIONS REQUIRED

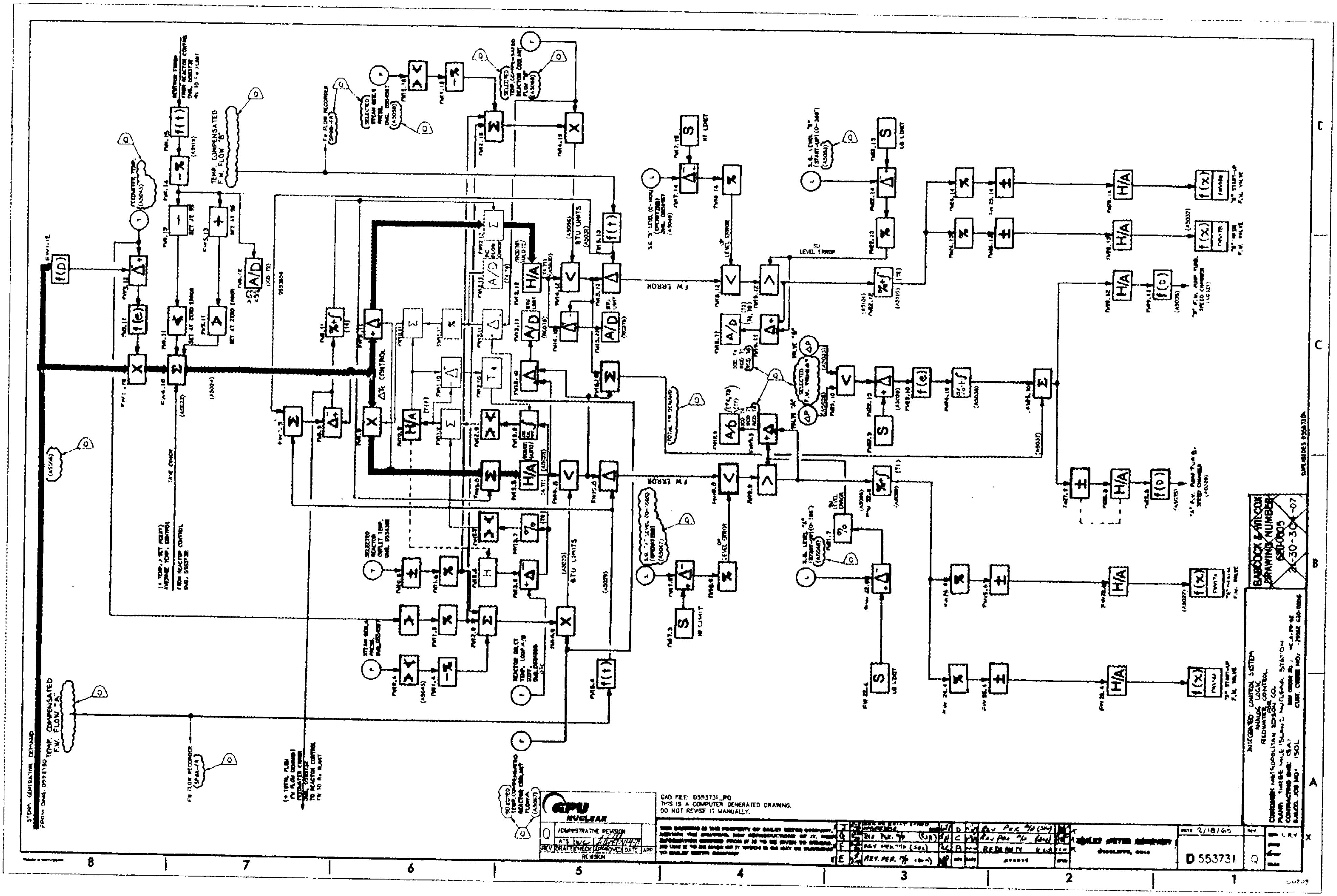
– **VERIFY** Automatic Action.

– **If** automatic feedwater control is inadequate, **then PLACE** feedwater in HAND IAW one or more of the following procedures:

  – OP-TM-621-471, ICS Manual Control. (for FW Loop Masters)

  – OP-TM-421-451, Manual Control of Feed Flow to A OTSG (for FW Valves)

  – OP-TM-421-452, Manual Control of Feed Flow to B OTSG (for FW Valves)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THIS IS A COMPUTER GENERATED DRAWING.
DO NOT REVISE IT MANUALLY.
GPU NUCLEAR
ADMINISTRATIVE REVISION
REV DRAFT CHECK APPROVED DATE APP
REVISION
D 553731
Q
FW ERROR
LEVEL ERROR
BTU LIMITS
TOTAL FW DEMAND
TEMP. COMPENSATED F.W. FLOW "A"
TEMP. COMPENSATED F.W. FLOW "B"
STEAM GENERATOR DEMAND
FEEDWATER TEMP.
ΔTc CONTROL
HI LIMIT
LO LIMIT
INTEGRATED CONTROL SYSTEM
METROPOLITAN EDISON CO.
THREE MILE ISLAND NUCLEAR STATION
BABCOCK & WILCOX
DRAWING NUMBER
8
7
6
5
4
3
2
1
A
B
C
D

Form ES-401-5 **Written Exam Question Worksheet** Q # 032

Examination Outline Cross-Reference Tier # 2

Evolution/System 011 Pressurizer Level Control Group # 2

K/A # K5.09 Page # 3.2-22 RO/SRO Importance Rating 2.6 2.7

Measurement Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: reason for manually controlling PZR level.

10CFR55.41(5)

10 CFR Part 55 Content ✔ 55.41 .5 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ☐ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- Pressurizer level control is selected to RC1-LT1.

Event:
- The following alarms actuate:
  - MAP G-1-5, Pzr Level Lo-Lo.
  - MAP G-3-5, Pzr Level Hi/Lo.
  - MAP D-3-1, MU Flow Hi.
- RC1-LT-3 and RC-LI-777 both indicate 235 inches, and rising.
- Makeup Tank level is 75 inches, lowering at a rate corresponding to the rise in Pressurizer level.

Based on these conditions identify the ONE selection below that describes why 1202-29 (first) immediate action guides the operator to transfer Pressurizer level control to HAND and then to adjust Makeup flow.

A. Stabilize pressurizer level by controlling Makeup Tank level steady.

B. Minimize decrease in RCP seal injection flow by reducing Makeup flow.

C. Prevent cavitation of the operating Makeup Pump by reducing Makeup flow.

D. Reduce RCS pressure rise to prevent automatic Pressurizer spray valve RC-V-1 operation.

Technical Reference 1202-29, Pressurizer System Failure, Step 2.2.1, Page 14, Rev. 60.

Open Exam Reference None.

Learning Objective V.D.11.02

Question Source ✔ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT in accordance with 1202-29 immediate action 2.2.1.

B INCORRECT in accordance with 1202-29 immediate action 2.2.1 statement.

Distracter is plausible because RCP seal injection flow is reduced temprorarily when Makeup flow is raised significantly.

C INCORRECT in accordance with 1202-29 immediate action 2.2.1 statement. Cavitation is a consideration under very high flow conditions where MU Tank level is significantly lower than the conditions presented.

Distracter is plausible because NPSH is reduced as pump flow raises, and reducing pump flow would raise NPSH.

D INCORRECT in accordance with 1202-29 immediate action 2.2.1 statement.

Distracter is plausible because RCS pressure would continue to rise to the spray valve setpoint if no compensatory actions were taken.

Comments None.

SECTION D

Malfunction In Pressurizer Level Indication Or Control

1.0 **SYMPTOMS**

1.1 Disagreement between pressurizer level indicators (computer and console) of more than 12 inches. RC1-LT1 (C1720), RC1-LT3 (C1722) and RC-LI-777.

1.2 Rapid change in indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction, of the pressurizer.

1.3 Possible high or low pressurizer level alarms.

- G-1-5, Pzr Level Hi-Hi
- G-2-5, Pzr Level Hi/Lo
- G-3-5, Pzr Level Lo-Lo

1.4 Pressurizer level indicator(s) **NOT** responding to changes in pressurizer level.

1.5 Hi makeup flow alarm (D-3-1, MU Flow Hi).

1.6 Pressurizer temperature fails to agree with saturation temperature for RCS pressure.

1.7 RCS pressure changes does **NOT** agree with PZR level changes.

2.0 **IMMEDIATE ACTION**

2.1 Automatic Action

2.1.1 If indication fails low

a. Pressurizer heaters trip at 80 inches.

b. Makeup valve MU-V-17 opens.

2.1.2 If indication fails high

a. Makeup valve MU-V-17 closes.

2.2 Manual Action

2.2.1 **TAKE** MU-V-17 under hand control AND **ADJUST** makeup flow to equal letdown flow minus seal injection to maintain makeup tank as constant as possible.

2.2.2 **SELECT** alternate pressurizer level transmitter.

2.2.3 **SELECT** alternate pressurizer temperature transmitter.

Form ES-401-5 **Written Exam Question Worksheet** Q # 034

Examination Outline Cross-Reference Tier # 2

Evolution/System 035 Steam Generator System (S/GS) Group # 2

K/A # K1.14 Page # 3.4-14 RO/SRO Importance Rating 3.9 4.1

Measurement Knowledge of the physical connections and/or cause-effect relationships between the S/GS and the following systems: ESF.

10 CFR Part 55 Content ✔ 55.41 .2 to .9 55.43

Proposed Question ☑ RO SRO ✔ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor operating at 100% power, with ICS in full automatic.
- HSPS Train B High RB Pressure enable/defeat switch is in DEFEAT for testing.

Event:
- OTSG 1A feedwater piping rupture inside the RB.

Current plant conditions:
- RB pressure 4.1 psig.
- OTSG 1A pressure is 590 psig and steady.
- OTSG 1B pressure is 1010 psig and steady.

Based on these conditions identify the ONE selection below that describes:
(1) OTSG level control setpoints.
(2) Controlling valves.

A. 1) OTSG 1A = 25 inches.
OTSG 1B = 25 inches.
2) OTSG 1A - EF-V-30A and EF-V-30D.
OTSG 1B - EF-V-30B and EF-V-30C.

B. 1) OTSG 1A = 0 inches.
OTSG 1B = 25 inches.
2) OTSG 1A = EF-V-30A and FW-V-16A.
OTSG 1B = EF-V-30C and FW-V-16B.

C. 1) OTSG 1A = 25 inches.
OTSG 1B = 25 inches.
2) OTSG 1A - EF-V-30A.
OTSG 1B - EF-V-30C and FW-V-16B.

D. 1) OTSG 1A = 0 inches.
OTSG 1B = 25 inches.
2) OTSG 1A - EF-V-30D and FW-V-16A.
OTSG 1B - EF-V-30B and FW-V-16B.

Technical Reference Lesson Plan 11.2.01.311, Heat Sink Protection System, PPT 76, Rev. 15.

Open Exam Reference None.

Learning Objective IV.E.05.04

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # QR4E05-04-Q07.

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because EF-V-30B/C will not get an open signal due to train B Hi RB Pressure enable defeat switch in defeat.

Distracter is plausible because 1) both OTSG level setpoints are correct and 2) if examinee did not take the Train B defeat into account then EF-V-30A/B/C/D would be controlling OTSG level.

B INCORRECT because OTSG 1A will not control at 0".

Distracter is plausible because 1) with Train B in defeat, EF-V-30D will not change its setpoint control from 0" and this may be incorrectly applied to EF-V-30A, 2) the valves for level control for OTSG 1B are correct.

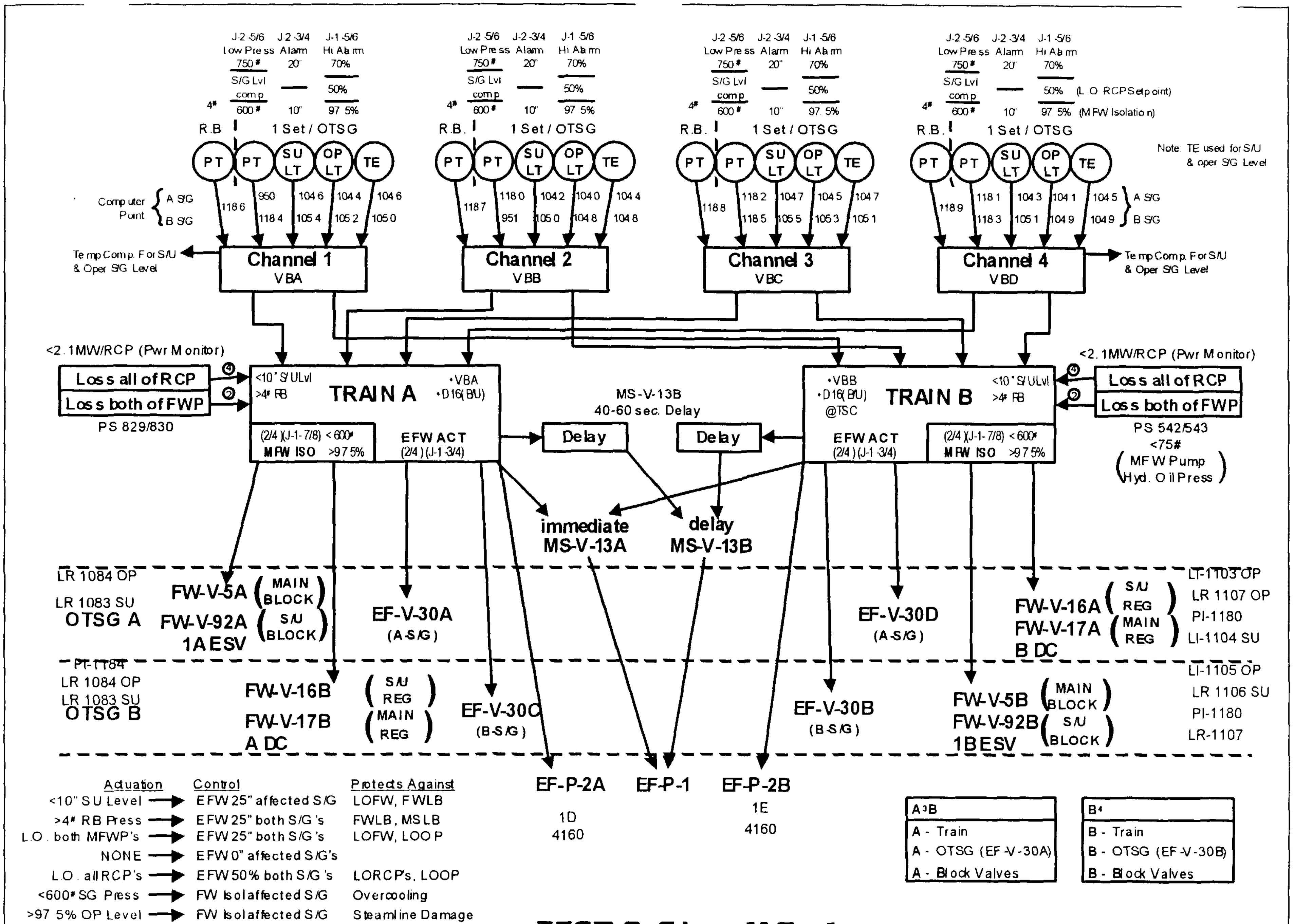
C CORRECT. Both OTSGs will be operate at 25" and OTSG 1B will be controlled by both EF-V-30B and FW-V-16B.

D INCORRECT because OTSG 1A will not control at 0 inches.

Distracter is plausible because 1) with Train B in defeat, EF-V-30D will not change its setpoint control from 0" and this may be incorrectly applied to EF-V-30A, 2) if a misconception exists about which EF-V-30s are controlled from each train then these would be the controlling valves. Also, FW-V-16B will act to control OTSG 1B level.

Comments Comparison to Audit exam question 59

- Both questions actuate EFW on 4# RB pressure signals.
- Audit question addresses EFW pump status, NRC question does not.
- Both questions address OTSG level controls.
- Audit question has both EFW trains operable, NRC question has Train B 4# EFW actuation bypassed.
- Both questions involve MFW line break.
- Audit question has 2 RCPs tripped, NRC question has all 4 RCPs running.
- Both questions have all 4 EF-V-30s in automatic.
- Audit question uses only EF-V-30s in the answer and distracters, NRC question also addresses MFW valves (FW-V-16s).
- Audit question uses level inputs to HSPS, NRC question uses OTSG pressure inputs to HSPS for a different EFW response.



HSPS Simplified

Form ES-401-5 **Written Exam Question Worksheet** Q # 033

Examination Outline Cross-Reference Tier # 2

Evolution/System 034 Fuel Handling Equipment System (FHES) Group # 2

K/A # A4.01 Page # 3.8-13 RO/SRO Importance Rating 3.3 3.7

Measurement Ability to manually operate and/or monitor in the control room: Radiation levels.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor is in refueling shutdown condition.
- Core re-load in progress.
- RM-G-7, Reactor Building Main Fuel Handling Bridge Radiation Monitor, scale selector switch is in the ALL position.

Event:
- CRO turns RM-G-7 scale selector switch to the 1E2 position, and then RELEASES the switch.

Identify the ONE selection below that describes current RM-G-7 Control Room indication.

A. 1E-1 to 1E7 mR/hr.

B. 1E-1 to 1E2 mR/hr.

C. 1E1 to 1E3 mR/hr.

D. 1E2 to 1E5 mR/hr.

Technical Reference Lesson Plan 11.2.01.118, Radiation Monitoring System, PPT 18, Rev. 18.

Open Exam Reference None.

Learning Objective IV.E.06.08

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the correct scale is 1E-1 to 1E2 mR/hr.

Distracter is plausible due to credible trainee misconception that this selector switch is spring return to the full scale indication position.

B CORRECT because this position selects 3 decades of indication - the highest being 1E2.

C INCORRECT because the correct scale is 1E-1 to 1E2 mR/hr.

Distracter is plausible because of credible misconception that this switch selects the middle decade for the 3-decade display.

D INCORRECT because the correct scale is 1E-1 to 1E2 mR/hr.

Distracter is plausible because of credible misconception that this switch selects the lowest decade for the 3-decade display.

Comments None.

Area Gamma Monitors

Control Panel Controls

- **Indication Selector Switches**
  - **OFF/ALL/10^7/10^6/10^5/10^4/10^3/10^2/CS**
  - **Read top scale in ALL (Lin-Log).**
  - **Read 3 decades in other positions (Lin-Log).**
- **Alarm Setpoint Indication**
  - **Depress Amber/Red Alert/Alarm Lights.**
- **Check Source Pushbuttons**
  - **RM-G-22, 23, 26, 27.**
- **Interlock Defeat Switches**
  - **RM-G-9, 16, 17, 18, 20, 21.**

Form ES-401-5 **Written Exam Question Worksheet** Q # 035

Examination Outline Cross-Reference Tier # 2

Evolution/System 041 Steam Dump System (SDS)/Turbine Bypass Control Group # 2

K/A # K4.01 Page # 3.4-24 RO/SRO Importance Rating 2.9 3.3

Measurement Knowledge of SDS design feature(s) and/or interlock(s) which provide for the following: RRG/ICS system.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Plant startup in progress.
- Reactor power 22%.
- Unit Load Demand 20%.
- Turbine Generator on line.
- Turbine Bypass Valves are closed in automatic.

Based on these conditions identify the ONE controlling setpoint for the Turbine Bypass valves.

A. 895 psig.

B. 960 psig.

C. 1010 psig.

D. 1040 psig.

Technical Reference Lesson Plan 11.2.01.055, Integrated Control System, Note on Page 23, Rev. 13.

Open Exam Reference None.

Learning Objective IV.E.27.17

Question Source ☐ New ✔ Bank ☐ Modified Bank Question # QR4E27-17-Q03. Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the main turbine is on line and demand is > 15%, therefore the bias is + 75 psig.

Distracter is plausible because this would be the setpoint if the main turbine wasn't on line or ULD demand was < 15%.

B CORRECT. The turbine is not tripped and the ULD demand is >15% therefore the setpoint is 885 psig + 75 psig.

C INCORRECT because this is the setpoint for post reactor trip.

Distracter is plausible because this is one setpoint for the turbine bypass valves.

D INCORRECT because this is the setpoint for the Atmospheric Dump Valves in this mode.

Distracter is plausible because this is the correct setpoint for the Atmospheric Dump Valves. If there was a misconception about which set of valves is currently controlling, then this answer might be chosen.

Comments Closed TBVs in the stem to eliminate ambiguity identified during exam validation.

2) For purpose of discussing the basic operation, assume that the reactor is in a hot-shut down (1 percent $\Delta k/k$ SD, Tave = 525°F)

   a) Bypass valves open maintaining 885+10 psig)

   b) Now – reactor is taken critical into the power range producing sensible heat

3) Turbine header pressure and SG header pressure (P_{HDRA} and $_{B}$) will increase

4) This will create a pressure error (P_{HDRA}-P_{HSP})

5) The error is sent to the bypass valve controller resulting in the opening of the bypass valves.

6) This reduces the SG header pressure.

7) As the error decreases, the valves begin to close until SG pressure equals the setpoint.

NOTE: At 15 percent demand and the turbine loaded, the bypass valves will be closed. At 15 percent demand the bypass valves bias selector is switched to 75 lb bias for transient pressure relief. Should a reactor trip occur, a +125 lb bias will be selected. This bias upon a reactor trip will allow a reactor cooldown and limit Tave to approximately 555°F during the trip.

c. Atmospheric Dump Valves

1) For the turbine bypass valves to be functional, the main turbine condenser must be available. When <u>the condenser is lost, due either to loss of cooling water (less than 2 CW-P-s are running) or high condenser pressure (greater than 7" Hg absolute)</u>, the turbine bypass valves to the condenser fail shut. (Latched closed)

   The atmospheric exhaust valves, if in auto would be utilized for pressure control at setpoint +Bias or open at 1040 until the condenser was again available.

Form ES-401-5 **Written Exam Question Worksheet** Q # 036

Examination Outline Cross-Reference Tier # 2

Evolution/System 045 Main Turbine Generator (MT/G) System Group # 2

K/A # K5.01 Page # 3.4-29 RO/SRO Importance Rating 2.8 3.2

Measurement Knowledge of the operational implications of the following concepts as they apply to the MT/G System: Possible presence of explosive mixture in generator if hydrogen purity deteriorates.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor is operating at 100% power with ICS in full automatic.

Event:
- MAP L-1-6, Gen H2 Seal Oil Trouble alarm actuates.
  - Local Hydrogen Seal Oil System alarm: Machine Gas Pressure High.
  - Local Hydrogen Seal Oil System alarm: Machine Gas Purity Low.
- Generator hydrogen gas analyzer is verified to be operating correctly.
- Gas purity is 90%, and slowly lowering.

From the list below identify the ONE selection that describes the concern if hydrogen GAS PURITY continues to lower.

A. Contamination of the hydrogen seal oil.

B. Development of a flammable gas mixture.

C. Accelerated corrosion of Main Generator components.

D. Reduction in Generator gas pressure requiring a load reduction.

Technical Reference Lesson Plan 11.2.01.433, PPT 9, Rev. 2.

Open Exam Reference None.

Learning Objective None.

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the concern is hydrogen in the generator becoming flammable as hydrogen concentration lowers towards 75% with oxygen mixing entering.

Distracter is plausible because contamination of the hydrogen seal oil would increase the corrosion rate of components in the system.

B CORRECT. Hydrogen concentrations between 4% and 75% are flammable when mixed with air.

C INCORRECT because the concern is hydrogen in the generator becoming flammable as hydrogen concentration lowers towards 75% with oxygen mixing entering.

Distracter is plausible because higher oxygen levels in the generator would raise corrosion rates.

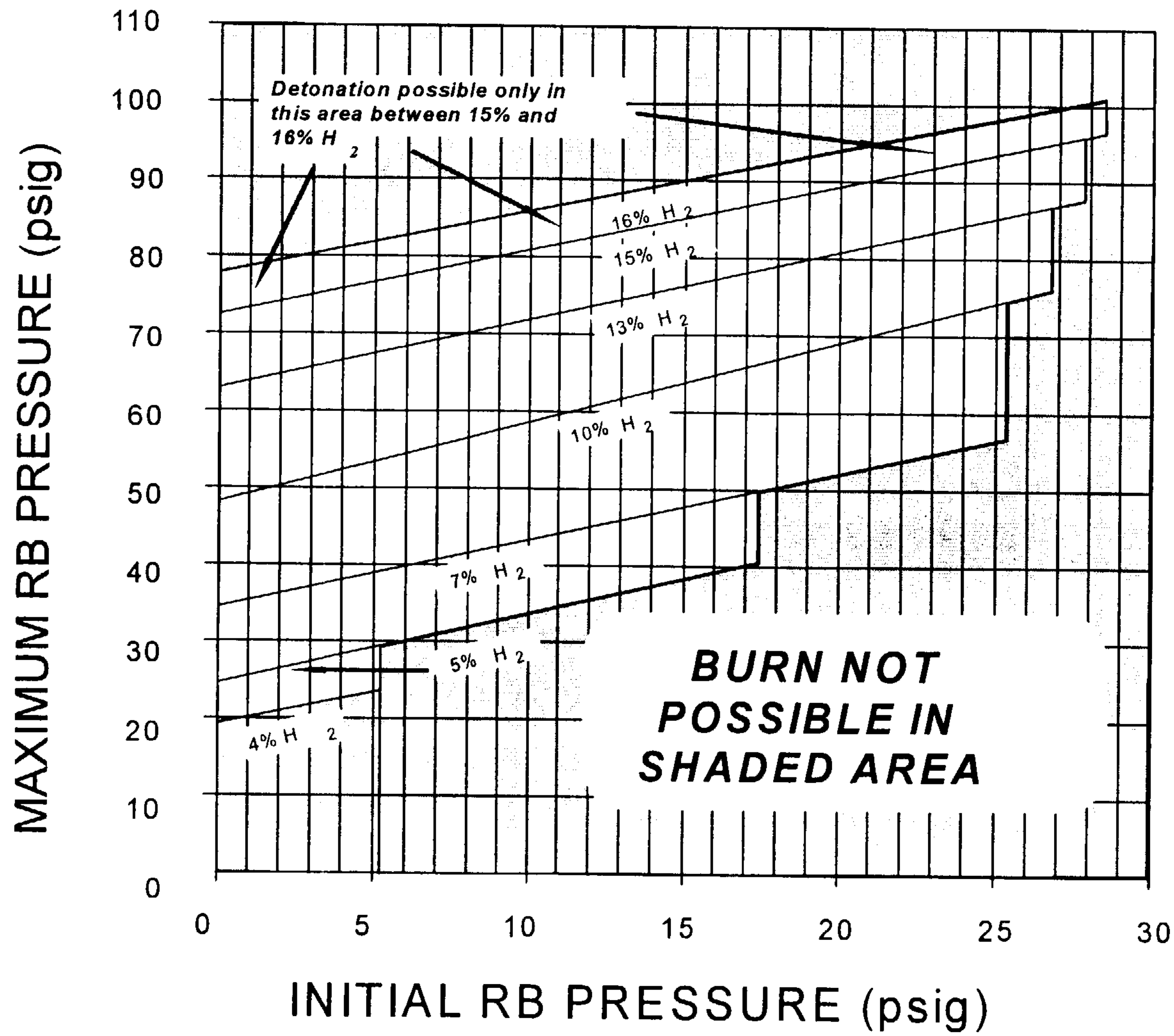
D INCORRECT because the concern is hydrogen in the generator becoming flammable as hydrogen concentration lowers towards 75% with oxygen mixing in. While load reduction would be required due to SO-

1-1 Machine Gas Pressure Lo alarm, it would not address the purity concern.

Distracter is plausible because lower hydrogen pressure would require a load reduction IAW OP-TM-301-000.

Comments None.

RB H_2 FLAMMABILITY LIMITS



Form ES-401-5 **Written Exam Question Worksheet** Q # 037

Examination Outline Cross-Reference Tier # 2

Evolution/System 055 Condenser Air Removal System (CARS) Group # 2

K/A # K3.01 Page # 3.4-33 RO/SRO Importance Rating 2.5 2.7

Measurement Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 55.43

Proposed Question ✔ RO SRO PRA Related **Correct Answer** A.

Plant conditions:
- Reactor at 100% power, with ICS in automatic EXCEPT ULD (in HAND).
- RCS boron concentration is 700 ppm.
- Operator completed adding 100 gallons of water to the Makeup Tank 10 minutes ago.
  - The batch calculation, blending water from two RC Bleed Tanks, intended to match RCS boron concentration.

Event:
- Automatic control rod withdrawal raises reactor power to 101%.
- Loop A and Loop B FW flows increase, corresponding to the reactor power increase.
- Main Generator output is reducing at 2 MW per minute.
- RCS Tave is constant at 579 degrees F.

Identify the ONE selection below that describes how all these indications could be true for one single event.

A. Increase in Main Condenser back pressure due to air in-leakage.

B. Increase in Main Generator MVARs due to grid voltage reduction.

C. Reactivity management event due to inadvertent control rod withdrawal.

D. Reactivity management event due to inadvertent RCS boron concentration increase.

Technical Reference Lesson Plan 11.2.01.055, ICS, Pages 21 and 22, Rev. 13.

Open Exam Reference None.

Learning Objective IV.D.4.12

Question Source ✔ New Bank Question #

Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. The events in the stem indicate a loss of plant efficiency due to high Main Condenser backpressure.

B INCORRECT because a rise in MVARS would not reduce generated MWs.

Distracter is plausible because a rise in MVARS would result in raising reactor power and feed water flow.

C INCORRECT because an inadvertent control rod withdrawal would raise RCS Tave and the stem states that Tave is held constant.

Distracter is plausible because a control rod withdrawal would result in the reactor power and feed flow rise given in the stem.

D INCORRECT because a small inadvertent RCS boron concentration rise would not cause reactor power to rise but rather a withdrawal of control rods to maintain RCS Tave at 579 degrees F.

Distracter is plausible because a RCS boron increase would result in an automatic control rod withdrawal to maintain Tave at 579 degrees F.

Comments Condenser back pressure effect on condenser/overall plant efficiency links this question to the KA.

For this purpose, slow integral action is applied to the megawatt error, which will result in a ratio change between the steam generator/reactor demand and the unit load demand. Action is required from this calibrating loop to compensate for changes in turbine/generator efficiency, steam enthalpy or feedwater flow measurement errors.

c. The calibration from this integral can only be effective when the steam generator and reactor are capable of responding to its actions. During a load transient, a megawatt error will exist, so the action of this integral is **blocked** (Output is held constant) when the load is changing faster than 2 percent per minute for 10 seconds, and remains blocked for two minutes after the load change stops.

MW integral conditions for being clocked or bled

When the unit is in the tracking mode, this integral will also be blocked and **bleed** (output is bled to 0), except when the turbine is on manual in which case the integral controls pressure.

When the steam generators are unable to respond, either due to their control being assumed by level control, or when the reactor is on a minimum limit, the correction from the megawatt calibrating integral is bled.

When the SG/Rx master is in hand, the input to the MW calibrating integral is the error across this H/A station. This provides a bumpless transfer to auto.

d. Inputs to the MW Calibrating Integral

1) MW Error: ICS Integrated Mode

2) Turbine Header Pressure Error: Turbine in Manual

Error across Steam Generator / Reactor Master H/A station when it is in Manual Control. Dominate Signal

3. Turbine Bypass System:

   a. The bypass valves serve the following functions:

      1) Provides pressure control at low loads before the turbine is capable of accepting pressure control (0-15 percent power).

      2) Provide a high pressure relief if the OTSG pressure exceeds its setpoint by 75 PSI. If the bypass valves were not biased to a higher setpoint, there would be two systems trying to control pressure which could lead to undesirable interaction between the two systems and thus give unstable pressure control.

         (Normal operations between 15 and 100 percent power)

      3) Provide pressure control after a reactor trip to prevent excessive cooling of the reactor coolant fluid.

      4) Provide an independent high-pressure relief that will operate proportionally to steam generator outlet pressure, if in auto (setpoint 1040)

   b. Basic operations of Bypass Valves

      1) The bypass controller receives its signal from the difference between (consider loop "A") the SG outlet pressure (P_{HDRA}) and the biased turbine header pressure setpoint (P_{HSP})

Form ES-401-5 **Written Exam Question Worksheet** Q # 038

Examination Outline Cross-Reference Tier # 2

Evolution/System 071 Waste Gas Disposal System (WGDS) Group # 2

K/A # A3.03 Page # 3.9-7 RO/SRO Importance Rating 3.6 3.8

Measurement Ability to monitor automatic operation of the Waste Gas Disposal System including: Radiation monitoring system alarm and actuating signals.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✓ 55.41 .7 ☐ 55.43

Proposed Question ✓ RO ☐ SRO ✓ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- RB Purge in progress for Containment Building pressure control.

Event:
- Significant FUEL PIN LEAK develops.
- RM-G-20 (RCDT Discharge Monitor) HIGH ALARM actuates.

Based on these conditions identify the ONE statement below that describes components affected during this situation.

A. AH-V-1A, 1B, 1C and 1D, Purge Isolation Valves, all close.

B. WDG-V-3, RB Vent Header Containment Isolation Valve, closes.

C. MU-V-2A, 2B, Letdown Cooler Outlet Isolation Valves, both close.

D. WDL-V-534, RB Sump Drain to Auxiliary Building Sump Valve, closes.

Technical Reference MAP C-1-1, RM-G-20, Page 36, Rev. 33.
Lesson Plan 11.2.01.118, Radiation Monitoring System, Page 28 and PPT 20, Rev. 18.

Open Exam Reference None.

Learning Objective IV.E.06.04

Question Source ☐ New ✓ Bank Question # AL4E06-04-Q08.

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ✓ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because these valves are interlocked with RM-A-9, which has not alarmed.

Distracter is plausible because with a RB purge in progress, automatic securing of the purge is a logical action.

B CORRECT. WDG-V-3 is interlocked with RM-G-20.

C INCORRECT because MU-V-2A/B are interlocked with RM-L-1, which has not alarmed.

Distracter is plausible because a significant fuel pin leak will eventually lead to RM-L-1 alarming, shutting MU-V-2A/B to minimize rad levels in the Auxiliary Building.

D INCORRECT because WDL-V-534 is interlocked with RM-G-9, which has not alarmed.

Distracter is plausible because WDL-534 provides a path outside containment if the RCDT rupture disk were to fail on high pressure.

Comments None.

C-1-1
Revision 33

ALARM:

RM-G-20 RC DRAIN TANK

SET POINTS:

Refer to Operating Procedure 1101-2.1 RMS setpoints.

CAUSES:

Fuel damage (Hi Activity in RCS) coupled with RCS leakage to drain tank.

AUTOMATIC ACTION:

Closes WDL-V-303, WDL-V-304, WDG-V-3, WDG-V-4.

OBSERVATION (CONTROL ROOM):

1. RM-G-20 "Alert (Warn)" Alarm on PRF.
2. RM-G-20 "High" Alarm on PRF.
3. RM-G-20 Indication on PRF > setpoints.

MANUAL ACTION REQUIRED:

1. Verify WDL-V-303, WDL-V-304, WDG-V-3, WDG-V-4 close.
2. Refer to Emergency Procedure 1202-11 (Hi RCS Activity); Emergency Procedure 1202-29 (Pressurizer System Failure).
3. Refer to EP 1202-12, Excessive Radiation Levels.

2) RM-G-9 HIGH ALARM closes dampers, AH-D-120, 121, 122. (between Fuel Handing Building Spent Fuel Pools isolating Unit I from Unit II) and trips AH-E-10.

3) RM-G-16 shuts CA-V-4A/5A (isolates A OTSG sample line).

4) RM-G-17 shuts CA-V-4B/5B (isolates B OTSG sample line).

5) RM-G-18 shuts CA-V-1/2/3/13 (isolates RCS sample line).

6) RM-G-20 shuts WDG-V-3/4 and WDL-V-303/304 (isolates gaseous and liquid [respectively] discharge from the RCDT).

7) RM-G-21 shuts WDL-V-534/535 (isolates RB sump from Aux. Bldg. Sump).

Area Gamma Monitors Interlocks

- RM-G-18-Close CA-V-1,2,3,13-(Primary sample lines)
- RM-G-20-Close WD-L-V-303/304 and WD-G-3/4 (isolate RC Drain Tk.)
- RM-G-21-Closes WD-L-V-534/535 (secures draining of R.B. sump to Aux. Bldg sump)

Form ES-401-5 **Written Exam Question Worksheet** Q # 039

Examination Outline Cross-Reference Tier # 1

Evolution/System 007 Reactor Trip Group # 1

K/A # EA2.04 Page # 4.1-3 RO/SRO Importance Rating 4.4 4.6

Measurement Ability to determine or interpret the following as they apply to a reactor trip: If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.

Event:
- Turbine trip due to high main condenser pressure.
  - All main turbine stop valves closed.
- Operator depressed the "Reactor Trip" and "DSS" pushbuttons.

Current conditions:
- Reactor power is 25% and slowly lowering.
- All 3 EFW Pumps are operating.
- OTSG atmospheric dump valves and some main steam safety valves are open.
- RCS Pressure has stabilized at 2350 psig.

Based on these conditions identify the ONE selection below that describes proper implementation of OP-TM-EOP-001, Reactor Trip.

A. Immediately initiate HPI, and then continue on to perform the next procedure action step.

B. IAAT RCS Pressure exceeds 2500 psig, initiate HPI, and then continue performing procedure steps.

C. Immediately initiate HPI, and then hold further procedure action(s) until reactor shutdown is confirmed.

D. IAAT power stabilizes at >5%, initiate HPI, and then hold further procedure action(s) until reactor shutdown is confirmed.

Technical Reference OP-TM-EOP-001, Reactor Trip, Page 1, Step 2.2 RNO.
OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 3.14, Page 6, Rev. 10.

Open Exam Reference None.

Learning Objective V.E.13.02

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the second part of the answer says to proceed to the next step. OP-TM-EOP-001 directs the operators to hold until the reactor is SHUTDOWN (<7% power).

Distracter is plausible because the first part of the answer (immediately initiate HPI) is correct.

B INCORRECT because HPI is initiated with RCS pressure < 2500 psig.

Distracter is plausible because EOP-001 result not obtained uses < 2500 psig as a condition for initiating HPI.

C CORRECT. EOP-001 states that when RCS pressure is < 2500 psig, initiate HPI, proceed once reactor is shutdown.

D INCORRECT because the distracter refers to an If At Any Time condition. EOP-1, Results Not Obtained, does not refer to an IAAT condition. The condition that the reactor is not shutdown with the reactor trip and DSS pushbuttons actuated is sufficient to trigger the results not obtained actions. Also, the >5% does not indicate that the reactor is not shutdown IAW the OS-24 definition of power <7%.

Distracter is plausible because the actions stated are correct if the reactor did not shutdown.

Comments None.

REACTOR TRIP

1.0 ENTRY CONDITIONS

- Any unplanned condition requiring an automatic or manual trip signal.
- A symptom of core cooling upset occurs while shutdown prior to DHR operation.

2.0 IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 2.1 **TRIP** the Reactor by depressing the "Reactor Trip" and "DSS" pushbuttons.	
_____ 2.2 **VERIFY** the reactor is shutdown.	_____1. **If** Main FW is **not** available, **then** _____ **ENSURE** Main Turbine is tripped _____ **ENSURE** EFW is actuated _____2. **MAINTAIN** primary-to-secondary heat transfer. _____3. **When** RCS pressure < 2500 psig, **then INITIATE** HPI. _____4. **When** the reactor is shutdown, **then CONTINUE**
_____ 2.3 **TRIP** the Turbine.	
_____ 2.4 **VERIFY** the turbine stop valves are closed.	_____ **PLACE** EHC-P-1A **and** EHC-P-1B in pull-to-lock. _____ **OPEN** EHC-V-FV1 (TB 305', EHC bypass valve at EHC pump skid).

3.13 PRIMARY-TO-SECONDARY HEAT TRANSFER (PSHT):

PSHT is the removal of sensible heat from the RCS to one or both OTSG(s). PSHT can be confirmed if all of the following conditions exist:

- Either OTSG has water level control and pressure control.
- RCS T_c is approximately the same as secondary T_{sat} and responds to changes in OTSG pressure.
- RCS forced or verified natural circulation is present.

3.14 REACTOR SHUTDOWN:

Heat generation by a self sustaining fission process has been effectively stopped. This reactor condition can be assessed immediately following reactor trip by Power Range Nuclear Instrumentation < 7 % FP. Nuclear Instrumentation may not represent reactor power during Loss of Coolant Accidents.

3.15 SYMPTOM CHECK:

A review of plant conditions to determine if a symptom of a core cooling upset exists. Refer to Attachment D, Symptom Check Guidelines.

3.16 TRICKLE FEED:

Trickle Feed is feeding a OTSG that cannot hold pressure (i.e., unisolable steam leak). RCS temperature and cooldown rate are controlled by feedwater flow instead of OTSG pressure.

3.17 VERIFY:

Observe whether a condition exists. No action is intended.

Exception: ENSURE and VERIFY in Emergency (1202 series) and Abnormal (1203 series) procedures are used interchangeably until they are revised.

Form ES-401-5 **Written Exam Question Worksheet** Q # 040

Examination Outline Cross-Reference Tier # 1

Evolution/System 008 Pressurizer Vapor Space Accident (Relief Valve Stu Group # 1

K/A # AA2.04 Page # 4.2-9 RO/SRO Importance Rating 3.2 3.4

Measurement Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: High temperature computer alarm and alarm type.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions;
- Reactor is operating at 100% power with ICS in full automatic.
- Parameters indicate an increase of 0.25 gpm leakage into the RC Drain Tank.
- RCS pressure steady at 2155 psig.
- Ambient temperature condition at RC-RV-2 PORV tailpipe is 100 degrees F.

Based on these conditions identify the ONE selection below that describes the operation and sensitivity of computer alarm A0517, RC-RV-2 TAILPIPE DELTA TEMP set at 30 degrees F.

(1) If hot fluid is flowing from the PORV, A0517 will alarm at an actual tailpipe temperature of ____________.
(2) This system (is/is NOT) sensitive enough to alarm with this amount of leakage.

A. (1) 130 degrees F.
   (2) Sensitive enough to alarm on this amount of leakage.

B. (1) 618 degrees F.
   (2) Sensitive enough to alarm on this amount of leakage.

C. (1) 130 degrees F.
   (2) NOT sensitive enough to alarm on this amount of leakage.

D. (1) 618 degrees F.
   (2) NOT sensitive enough to alarm on this amount of leakage.

Technical Reference 1105-10A, Plant Computer Attributes, Points A0517, 518, 519, Pages E2-40 and E4-27, Rev. 50.

Open Exam Reference None.

Learning Objective IV.A.01.13

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer.

B INCORRECT because alarm will actuate at 130 degrees F.

Distracter is plausible because temperature (618) is 30 degrees delta-T from Pressurizer steam temperature in stem, and system sensivity will result in alarm actuation with this amount of leakage.

C INCORRECT because system design and sensitivity supports the alarm function for this amount of leakage.

Distracter is plausible because the temperature listed for the alarm is correct, and the examinee may think this amount of leakage is below "MDA."

D INCORRECT because alarm will actuate at 130 degrees F and system design and sensitivity supports the alarm function for this amount of leakage.

Distracter is plausible because temperature (618) is 30 degrees delta-T from Pressurizer steam temperature in stem, and the examinee may think this amount of leakage is below "MDA."

Comments None.

A05 PRESSURIZER/ RCS INVENTORY

POINT NO	DESCRIPTOR	UNITS	POINT SOURCE	SCAN	ALM COND	ALM PRI	ALARM LIMITS	ALARM BASIS	ALM DEADB	ALARM CUTOU	CUTOU T STATE	REFERENCE DOCUMENT
A0402	PORV POSITION (SONIC FLOW SENSOR		VMS ACC 1/2	30	ON	2	HI 9.0	0006B	1.0	NONE	N/A	OP-TM-MAP-G0107
A0408	RC VALVE RV2 FLOW (IN.H2O)	INH2O	RC-DPT-921	30	ON	2	HI 20	0006C	5.0	NONE	N/A	OP-TM-MAP-G0106
A0409	RC VALVE RV1A FLOW (IN.H2O)	INH2O	RC-DPT-922	30	ON	2	HI 20	0006C	5.0	NONE	N/A	OP-TM-MAP-G0106
A0410	RC VALVE RV1B FLOW (IN.H2O)	INH2O	RC-DPT-923	30	ON	2	HI 20	0006C	5.0	NONE	N/A	OP-TM-MAP-G0106
A0459	RC DRAIN TANK WDL-T-3 TEMP	DEGF	WDL-TE-605	30	ON	3	HI 120	0004D	2.0	NONE	N/A	
A0466	RC LOOP A HOT LEG LEVEL	FT	RC-LT-1033	30	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0467	RC REACTOR VESSEL LEVEL A	FT	RC-LT-1035	30	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0468	RC LOOP B HOT LEG LEVEL	FT	RC-LT-1034	30	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0469	RC REACTOR VESSEL LEVEL B	FT	RC-LT-1036	30	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0501	RC PRESSURIZER LEVEL 1 - INST DP	INH2O	RC1-LT-1	05	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0503	RC PRESSURIZER LEVEL 3 - INST DP	INH2O	RC1-LT-3	05	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0504	RC PRESSURIZER TEMP	DEGF	RC2-MS	30	ON	3	LO 620	0003B	2.0	L2951	COLD	
A0516	RC PRESSURIZER SURGE LINE TEMP	DEGF	RC9-TE	60	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A0517	RC-RV2 TAILPIPE DELTA TEMP	DEGF	RC10-TE1AP	60	ON	3	HI 30	0006D	3.0	NONE	N/A	OP-TM-PPC-A0517
A0518	RC-RV1A TAILPIPE DELTA TEMP	DEGF	RC10-TE2AP	60	ON	3	HI 30	0006D	3.0	NONE	N/A	OP-TM-PPC-A0518
A0519	RC-RV1B TAILPIPE DELTA TEMP	DEGF	RC10-TE3AP	60	ON	3	HI 30	0006D	3.0	NONE	N/A	OP-TM-PPC-A0518
A0520	PRESSURIZER SPRAY LINE TEMP	DEGF	RC11-TE	60	ON	3	LO 425	0004E	2	L2951	COLD	
A0835	RC DRAIN TANK LEVEL	FT	WDL-LT-115	30	ON	3	LO 7.46 HI 8.08	N/A	N/A	L2952	ON	ANN LWDS-1-5
A1029	RC PRESSURIZER LVL (FROM LT-777)	INH20	RC-LT-777	05	OFF	0	NONE	N/A	N/A	NONE	N/A	N/A
A5059	PRESSURIZER LEVEL /SELECTED	IN	RC1-LT-1/3	01	ON	2	HI 315 LO 20	0001C	5.0	NONE	N/A	
C1716	SATURATION PRESSURE FOR PZR TEMP	PSIG	A0504	60	OFF	0	NONE		N/A	NONE	N/A	
C1720	RC PRESSURIZER LEVEL 1 - COMP	IN	A501 & A504	15	ON	0	HI 240 LO 200	0005B	2.0	L2951	COLD	

ALARM BASIS NO. 0006A
The high alarm warns the operator that procedural guidance must be followed to protect CRD equipment. If CRD cooling water outlet temperature reaches 160°F, MU-V-1A/B close, isolating Letdown and providing more cooling to the CRD stators. The HI2 alarm alerts the operator that the CRD in question must be deenergized. If more than one stator temperature reaches 180°F the reactor must be tripped (EP 1202-08, CRD Equipment Failures - CRD Malfunction Action).

ALARM BASIS NO. 0006B
The high alarm provides gross open/shut indication for the PORV.

ALARM BASIS NO. 0006C
The high alarm provides gross open/shut indication for the PORV and Code Safety valves.

ALARM BASIS NO. 0006D
The delta-T instrument provides sensitive indication of flow through the pressurizer code safety valves or power operated relief valve. The alarm setpoint was chosen such that any deviation from normal would be alarmed. The high alarm will provide indication of hot fluid flow through the pipe (OP 1103-5, Pressurizer Operations).

ALARM BASIS NO. 0006E
The high alarm warns operators that RCS pressure is within 50 psig of the PORV setpoint (RCS temperature > 275°F). Operators should quickly evaluate and take corrective actions as necessary.

The low alarm warns operators that the ESAS setpoint has been reached with a safety grade instrument independent of ESAS. If the RCS is < 1600 psig ESAS must be verified/initiated, and an Unusual Event declared.

The Lo2 alarm warns operators that the backup ESAS setpoint is being approached (500 psig), and Core Flood will be discharging if CF-V-1A/B are not closed.

References:
1. TMI-1 Tech. Spec. 3.1.12.
2. TMI-1 Tech. Spec. 3.5.3.
3. OP-TM-EOP-001, Reactor Trip.

ALARM BASIS NO. 0006F
The high alarm setpoint is approximately 0.1 Mlb/hr above normal flow noise peaks at full power. High flow indicates a decrease in cold leg temperature, a mismatch in RCS loop flows, or an instrument problem. Operators should investigate, and take corrective actions as necessary.
The low alarm gives early warning that low flow noise peaks are resulting in RPS Power/Flow/Imbalance trip setpoints below approximately 104% power. This provides ample margin to avoid trips during anticipated transients.

References: TMI-1 Tech. Spec. 2.3, Figure 2.1.3.

Form ES-401-5 **Written Exam Question Worksheet** Q # 041

Examination Outline Cross-Reference Tier # 1

Evolution/System 009 Small Break LOCA Group # 1

K/A # EA2.10 Page # 4.1-5 RO/SRO Importance Rating 3.1 3.7

Measurement Ability to determine or interpret the following as they apply to a small break LOCA: Airborne activity.

10CFR55.41(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:

- Reactor tripped from full power.
- RCS pressure 1680 psig, lowering at 10 psig per minute.
- NO ES Actuations.
- Pressurizer level 85 inches, lowering at 1 inch per minute.
- Makeup valve MU-V-17 full open.
- Makeup Tank level lowering at 5 inches per minute.
- RCP labyrinth seal dP indications normal.
- Core exit incore thermocouple temperatures 535 degrees F, lowering at 1 degree/minute.
- RM-A-2 containment airborne radiation levels are rising slowly.
- Containment pressure 1.5 psig rising at 0.1 psig per minute
- Startup FW flow indicated to both OTSGs.
- OTSG 1A level 25 inches, steady.
- OTSG 1B level 27 inches, rising slowly.
- OTSG pressures both 950 psig, lowering at 10 psig per minute.

Based on these conditions identify the ONE selection below that describes the event in progress inside the Containment Building.

A. Small break LOCA.

B. OTSG 1B tube rupture.

C. OTSG 1A FW line break.

D. Combined RCP seal #1 leak-off line leak.

Technical Reference OP-TM-EOP-001 Step 4.3, Page 7, Rev. 5.

Open Exam Reference None.

Learning Objective V.E.13.11

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. Conditions represent initial symptoms of small break LOCA inside containment.

B INCORRECT because there is no indication of secondary plant activity increase.

Distracter is plausible this presents a possible flowpath for RCS inventory loss, and stem gives rising level in OTSG 1B.

C INCORRECT because the a secondary steam leak would not produce rising RB activity.

Distracter is plausible because a FW line break inside containment would produce all the symptoms in the stem except rising RB activity.

D INCORRECT because a seal return line rupture is a low energy RCS leak that would neither raise RB pressure significantly with normal containment cooling systems operating nor lower RCS pressure to actuate ES.

Distracter is plausible because it is supported by all the symptoms of RCS inventory loss and rising RB activity.

Comments None.

4.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ Time	
_____ 4.1 **ENSURE** performance of an alarm review.	
_____ 4.2 **REQUEST** SM evaluate Emergency Action Levels (EALs).	
_____ 4.3 **VERIFY** RCS inventory can be controlled by Normal Make Up (shift management may direct entry into OP-TM-EOP-006 with smaller leaks).	_____ **GO TO** OP-TM-EOP-006.
_____ 4.4 **ENSURE** all Reactor Trip Isolation valves are closed.	
_____ 4.5 **VERIFY** containment pressure < 2 psig.	_____ **INITIATE** Guide 18.
_____ 4.6 **INITIATE** OP-TM-301-151, Main Turbine Generator Operating Mode to Standby Mode.	
_____ 4.7 **INITIATE** Attachment 1 (Align plant equipment for Hot Shutdown).	
_____ 4.8 **INITIATE** event notification per 1044, "Event Review and Reporting Requirements" (NRC notification is required within 4 hours of event).	
_____ 4.9 **VERIFY** at least one RCP is operating.	_____ **INITIATE** Guide 7 to start a RC Pump.
_____ 4.10 **INITIATE** AP 1063, "Reactor Trip Review Process".	
_____ 4.11 **INITIATE** 1102-4 Enclosure 2C "Actions following a power reduction"	
_____ 4.12 **INITIATE** 1102-10 Enclosure 1B "Actions following a Plant Shutdown".	

Form ES-401-5 **Written Exam Question Worksheet** Q # 042

Examination Outline Cross-Reference Tier # 1

Evolution/System 011 Large Break LOCA Group # 1

K/A # EA1.10 Page # 4.1-7 RO/SRO Importance Rating 4.1 3.8

Measurement Ability to operate and monitor the following as they apply to a Large Break LOCA: AFW and SWS pumps.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ✔ 55.41 .7 ☐ 55.43

Proposed Question ✔ RO ☐ SRO ✔ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor trip due to loss of coolant inside the RB.
- RCS subcooled margin 5 degrees F.
- Both Core Flood Tanks discharging water into the RCS.
- RB pressure 37 psig, rising at 1 psig per minute.
- All 4 RCPs tripped.
- Train A and Train B LPI flow is 2800 gpm EACH.
- OTSG 1A and OTSG 1B levels 45%, rising at 1% per minute.

Based on these conditions identify the ONE selection below that describes operational requirements as described in procedures for:
(1) Emergency Feedwater Pumps.
(2) Nuclear River Water Service Pumps.

A. (1) Operation of EFW pumps is NOT required.
(2) 3 NR Pumps are required to operate.

B. (1) If possible operate all 3 EFW pumps.
(2) 3 NR Pumps are required to operate.

C. (1) Operation of EFW pumps is NOT required.
(2) If possible operate 2 NR Pumps.

D. (1) If possible operate all 3 EFW pumps.
(2) If possible operate 2 NR Pumps.

Technical Reference OP-TM-EOP-010 Guide 15.1, Return EFW to Standby, Page 31, Rev. 3.
TS 3.3.1.4.b, Cooling Water Systems, Page 3-22, Amendment 227.
TS 3.3.2, Page 3-23, Amendment 229.
1202-38, Nuclear Services River Water Failure, Page 2, Rev. 40.

Open Exam Reference None.

Learning Objective IV.C.05.06

Question Source ✔ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✔ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because EFW operation is required in accordance with Rule 1 and does NOT meet return to standby criteria defined in Guide 15.1, and because station design prevents operation of 3 NR Pumps during ES conditions (non-ES selected pump is locked out).

Distracter is plausible because of EFW is not required to augment RCS break cooling for large break LOCA conditions, and Tech Specs normally requires 2 NR Pumps to be operable when the reactor is critical.

B INCORRECT answer because station design prohibits operation of 3 NR Pumps during ES conditions (non-

ES selected pump is locked out).

Distracter is plausible because because EFW operation is required in accordance with Rule 1 and does NOT meet return to standby criteria defined in Guide 15.1, and normally (Non-ES) all three NR Pumps are available to operate.

C INCORRECT answer because EFW is required in accordance with Rule 1 and does NOT meet return to standby criteria defined in Guide 15.1.

Distracter is plausible because because of EFW is not required to augment RCS break cooling for large break LOCA conditions.

D CORRECT answer because (1) EFW operation is required in accordance with Rule 1 and does NOT meet return to standby criteria defined in Guide 15.1, and (2) only 2 NR Pumps can be operated due to plant modification that trips and locks out the non-ES selected NR Pump to limit diesel generator loading.

Comments Question addresses knowledge of requirements for operation of the AFW and SWS pumps as applicable to a Large Break LOCA in accordance with EOP and Technical Specification requirements. This knowledge is prerequisite to monitoring and determining operational requirements during a LOCA, and therefore supports the KA (ability to operate and monitor...).

Guide 15.1
Return EFW to Standby

When ALL of the following conditions are satisfied,

_____ SCM > 25°F
_____ Main Feedwater flow has been established to each available OTSG
_____ At least one reactor coolant pump is operating
_____ OTSG level > 20" in each available OTSG.
_____ RB pressure < 2 psig
_____ CRS concurrence has been obtained

then PERFORM the following to place EFW in standby.

1. **PLACE** the EFW control valves in Manual

_____	EF-V-30A	_____	EF-V-30B
_____	EF-V-30D	_____	EF-V-30C

_____ 2. **ENSURE** all EFW actuation switches (8) are in DEFEAT.

_____ 3. **CLOSE** EF-V-30A & D **and ENSURE** OTSG A level is maintained with Main FW

_____ 4. **CLOSE** EF-V-30B & C **and ENSURE** OTSG B level is maintained with Main FW

_____ 5. **PLACE** Train A **and** Train B EFW Actuation switches for Loss of RCPs **and** High RB Pressure in ENABLE. (4 switches)

_____ 6. **If** at least one FW pump is RESET, **then PLACE** Train A **and** Train B EFW Actuation for Loss of FWPs in ENABLE (2 switches)

_____ 7. **If** OTSG A level > 20" and OTSG B level > 20", **then PLACE** Train A **and** Train B EFW Actuation for Lo-Lo OTSG Level in ENABLE (2 switches)

_____ 8. **PLACE** EF-P-2A in Normal-after-stop

_____ 9. **PLACE** EF-P-2B in Normal-after-stop

_____ 10. **ENSURE** MS-V-10A is CLOSED **and CLOSE** MS-V-13A

_____ 11. **ENSURE** MS-V-10B is CLOSED **and CLOSE** MS-V-13B

12. **PLACE** each EFW control valve in AUTO **and SELECT** REMOTE setpoint

_____	EF-V-30A	_____	EF-V-30B
_____	EF-V-30D	_____	EF-V-30C

b. **CFT** boron concentration shall not be less than 2,270 ppm boron. Specification 3.3.2.1 applies.

c. The electrically operated discharge valves from the **CFT** will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

d. DELETED

e. **CFT** vent valves CF-V-3A and CF-V-3B shall be closed and the breakers to the CFT vent valve motor operators shall be tagged open, except when adjusting core flood tank level and/or pressure. Specification 3.0.1 applies.

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The sodium hydroxide (NaOH) tank shall be maintained at 8 ft. $\pm$6 inches lower than the BWST level as measured by the BWST/NaOH tank differential pressure indicator. The NaOH tank concentration shall be 10.0 $\pm$.5 weight percent (%). Specification 3.3.2.1 applies.

c. All manual valves in the discharge lines of the **NaOH** tank shall be locked open. Specification 3.3.2.1 applies.

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.

b. Two nuclear service river water pumps must be OPERABLE.

c. Two decay heat closed cycle cooling water pumps must be OPERABLE.

d. Two decay heat river water pumps must be OPERABLE.

e. Two reactor building emergency cooling river water pumps must be OPERABLE.

3.3.1.5 Engineered Safeguards Valves and Interlocks Associated with the Systems in Specifications 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4 are OPERABLE. Specification 3.0.1 applies.

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS (Contd.)

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.*

3.3.2.1 If the CFT boron concentration is outside of limits, or NaOH tank is outside the limits of 3.3.1.3.b or any manual valve in the NaOH tank discharge lines are not locked open, restore the system to operable status within 72 hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both CFTs shall be OPERABLE at all times.

b. Both the motor operated valves associated with the CFTs shall be fully open at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be verified to be OPERABLE.

* **In accordance with AmerGen License Change Application dated February 14, 2001, and any requirements in the associated NRC Safety Evaluation, a portion of the Nuclear Service Water System piping between valves NR-V-3 and NR-V-5 may be removed from service and Nuclear Services River Water flow realigned through a portion of the Secondary Services River Water System piping for up to 14 days. This note is applicable for one time use during TMI Unit 1 Operating Cycle 13.**

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location and sump recirculation switchover setpoint. The limits on contained water volume, NaOH concentration and boron concentration ensure a pH value of

Form ES-401-5 **Written Exam Question Worksheet** Q # 043

Examination Outline Cross-Reference Tier # 1

Evolution/System 015/017 Reactor Coolant Pump (RCP) Malfunctions Group # 1

K/A # AA1.11 Page # 4.2-11 RO/SRO Importance Rating 2.5 2.4

Measurement Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP on/off and run indicators.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- RC-P-1A motor amps and temperatures rising.

Based on these conditions identify the ONE selection below that describes indications provided by RC-P-1A status lamps as motor current continues to rise to the automatic trip setpoint and then trips - specifically:
(1) When MAP F-1-2, RCP Motor Overload, actuates due to high motor current.
(2) Following RC-P-1A automatic breaker trip due to motor overload.

A. (1) Amber lamp will light.
(2) Green and Amber lamps (only) will be lit.

B. (1) Amber light will NOT light.
(2) Green and Amber lamps (only) will be lit.

C. (1) Amber lamp will light.
(2) Red, Green and Amber lamps will be lit.

D. (1) Amber lamp will NOT light.
(2) Red, Green and Amber lamps will be lit.

Technical Reference OP-TM-MAP-F0102, RCS Motor Overload, Rev. 0.
OP-TM-MAP-F0101, RCP Motor Trip, Rev. 0.

Open Exam Reference None.

Learning Objective IV.A.05.28

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer.

B INCORRECT answer because the amber lamp will be lit due to motor overload prior to tripping the breaker.

Distracter is plausible because is describes correct lamp status FOLLOWING breaker trip.

C INCORRECT answer because the red lamp will not be lit following breaker trip.

Distracter is plausible because is describes correct green and amber lamp status BEFORE and FOLLOWING breaker trip.

D INCORRECT answer because the amber lamp will be lit due to motor overload prior to tripping the breaker. Additionally, the red lamp will not be lit following breaker trip.

Distracter is plausible because is describes correct lamp status following breaker trip.

Comments None.

RCP MOTOR OVERLOAD	MAP F-1-2 System 226	**OP-TM-MAP-F0102** Revision 0 Page 1 of 1 **Level 2 – Reference Use**

1.0 **SETPOINTS**

- 115% of full load amps (115% = 780 amps) (51X relay)

2.0 **CAUSES**

- Mechanical failures: Shaft or seal binding, Bearing failure, Backstop damage
- Electrical failures: Insulation breakdown, high resistance connection
- Low Voltage 1A or 1B 7KV Bus.

3.0 **AUTOMATIC ACTIONS** - None

4.0 **MANUAL ACTIONS REQUIRED**

- **OBSERVE** the following:
  - RCP Ammeter (CC) – red band (115%) (normal ≈ 85% @ 579°F T_{AVE})
  - RCP Control switch (CC) - amber light Lit
  - Bentley-Nevada (PLF) - elevated vibrations
  - 1A / 1B 7KV Bus voltages (PR) – > 6.15 KV
- **DISPATCH** operator to affected RCP breaker to obtain RCP amps (normal 560 to 600 amps).
- **If** RCP Motor current at ≥ 780 amps at breaker indicator, **then PERFORM** OP-TM-226-150 series procedure to place affected RCP in the Standby mode.

Level 2 – Reference Use

1.0 SETPOINTS

- Breaker trip with C/S in Normal-After-Start position.

2.0 CAUSES

- 1A or 1B 6900V Bus voltage - ≤ 6.15 KV for ≈ 5 seconds
- Overcurrent - 1273 amps (delay) / 7478 amps (instantaneous)
- RCP Motor Phase differential > 5 amps
- Seal Injection < 22 gpm and IC flow <550 gpm for > 10 seconds.

3.0 AUTOMATIC ACTIONS

- Possible Reactor Trip (Ref G-1-1)
- Possible Plant Runback (Ref H-1-1)
- Tripped RCP HP lift pump starts.
- Reactor / turbine trip and EFW actuation on loss of all RCPs.

4.0 MANUAL ACTIONS REQUIRED

NOTE: With OTSG levels > LLLs, a Trip on one A loop pump will require a Feedwater re-ratio of ≈ 30% / 70% of flow to A / B OTSG. A trip of one B loop pump will require a re-ratio of ≈ 70% / 30% of Feedwater flow to A / B OTSG.

- **ENSURE** ICS runback and feedwater flow re-ratio.
- **DETERMINE** tripped RCP(s) from disagreement light(s) (CC) **and**:
  - **ENSURE** at least one RC-P-2 pump in service (Oil Lift)(CC).
  - **START** at least one RC-P-3 pump (Backstop Oil)(CC).
- **VERIFY** adequate FW Flow for primary heat removal .
- **If** both H-1-6 and H-1-7 (OTSG A/B LLLs) are Clear,
  then ENSURE ΔTC returned to ≈ 0 °F (RC-8 DTI)(CC).
- **PERFORM** OP-TM-226-150 series procedures to place tripped RCP(s) in Standby Mode.

Written Exam Question Worksheet

Examination Outline Cross-Reference Tier # 1

Evolution/System 022 Loss of Reactor Coolant Makeup **Group #** 1

K/A # AA1.01 **Page #** 4.2-13 **RO/SRO Importance Rating** 3.4 3.3

Measurement Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Pump Makeup: CVCS letdown and charging.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** A.

Initial conditions:
- Reactor operating at 100% power with ICS in full automatic.
- Intermediate Closed Cooling pump IC-P-1A operating.

Event:
- Makeup pump MU-P-1B trip.

Current conditions:
- RCS letdown flow isolated.
- RCP seal injection and RCS Makeup control valves closed.
- Crew is prepared to start MU-P-1A.
- Operator is directed by procedure to manually control components to gradually restore normal system configuration.

Based on these conditions identify the ONE selection below that describes two parameters the operator will MONITOR AND MANUALLY CONTROL in order to minimize thermal shock to equipment and systems while restoring normal system configuration.

A. (a) RCS makeup flow.
   (b) RCS letdown flow.

B. (a) RCP seal injection flow.
   (b) Intermediate Closed Cooling system flow.

C. (a) RCP seal #1 leak off flow.
   (b) Makeup pump recirculation flow.

D. (a) RCP Seal #2 leak off flow.
   (b) Decay Heat Closed Cooling system flow.

Technical Reference 1203-15, Loss of RC Makeup/Seal Injection, Page 4, Rev. 28.

Open Exam Reference None.

Learning Objective IV.A.09.52

Question Source ☑ New ☐ Bank **Question #**

☐ Modified Bank **Parent Question #**

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer. Refer to 1203-15.

B INCORRECT answer because Intermediate Closed Cooling system flow is not controlled by the operator for this reason.

Distracter is plausible because RCP Seal Injection flow is re-established gradually under these guidelines, and the operator is required to limit seal water temperature drop at the bearing to 1 degree per minute.

C INCORRECT answer because Makeup pump recirculation flow is not controlled by the operator for this reason.

Distracter is plausible because RCP Seal #1 leak off flow could be isolated for these conditions, and re-establishing this flow path could induce a thermal cycle on the equipment.

D INCORRECT answer because Seal #2 leak off flow and DCCS flow are not controlled by the operator for this reason.

Distracter is plausible because seal #2 flow changes when seal #1 leak off is isolated or re-established, and DCCS flow is controlled (under other circumstances) to limit RCS temperature changes.

Comments None.

NOTE

A comparison of changes in Make-up Tank level versus pressure can be used to validate instrument operation.

CAUTION

Ensure that the Make-up Pump has an adequate suction supply prior to starting the standby pump.

Initials

_____ f. Open/verify open MU-V-12 **OR** open BWST supply valve, MU-V-14A/B, if MU-V-12 cannot be opened or Make-up Tank inventory is questionable.

_____ g. Start standby make up pump and ensure cooling water supplied.

- MU-P-1B power can be swapped to 1D 4160V Bus IAW OP-TM-211-449.

_____ h. Slowly re-establish RCP Seal injection flow at a rate so seal water temperature at the bearing is decreasing less than 1°F/min, (Computer points A0521 to A0524).

_____ i. When a flow rate of 38 gpm is established place MU-V32 control station in AUTO.

_____ j. If isolated, restore pressurizer level gradually and return MU-V17 control station to AUTO.

_____ k. If isolated, restore letdown flow gradually IAW OP 1104-2 to minimize letdown cooler thermal shock.

6. Action for Makeup Pump Operating and MU-V17 failed closed

_____ a. Shift MU-V17 MU flow control station to manual and restore pressurizer level.

_____ b. If MU-V17 control station has failed, control pressurizer level with MU-V217 or local control of MU-V-92 (MU-V17 bypass valve).

_____ c. Isolate MU-V-17 by closing MU-V91A and B as time allows.

_____ d. If isolated, restore letdown flow gradually IAW 1104-2 to minimize letdown cooler thermal shock.

7. Action for Makeup Pump Operating and MU-V-32 Failed Fully or Partially Closed

_____ a. Attempt to restore seal injection by placing MU-V-32 control station in "Manual" and attempting to open MU-V-32 and/or starting an additional MU Pump (normally MU-P-1A).

_____ b. If MU-V-32 cannot be opened from the Control Room, have an AO slowly throttle open MU-V-90 to obtain desired seal injection flow rate and then isolate MU-V-32 by closing MU-V-89A and/or 89B.

Form ES-401-5 **Written Exam Question Worksheet** Q # 045

Examination Outline Cross-Reference Tier # 1

Evolution/System 025 Loss of Residual Heat Removal System (RHRS) Group # 1

K/A # AK2.03 Page # 4.2-16 RO/SRO Importance Rating 2.7 2.7

Measurement Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps.

10 CFR Part 55 Content ☑ 55.41 .8/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Initial plant conditions:
- Reactor at Cold Shutdown conditions.
- RCS pressure 50 psig, controlled by Pressurizer heaters.
- Decay Heat Removal (DHR) Train A is operating.

Activities in progress:
- Securing DHR cleanup flow through the Liquid Waste Disposal System.
- DH-V-1, DHR dropline isolation valve, surveillance test.
- Local venting of DH-P-1A pump casing.

Event:
- Electrical fault trips normal feeder breaker to 1R 480V Bus.

Current condition:
- Computer alarm indicates rising DH Suction temperature.

Based on these conditions, identify the ONE selection below that describes the cause for the rising DHR suction temperature.

A. Closure of DH-V-1.

B. Trip of 1R 480V Bus.

C. DH-P-1A pump venting.

D. Securing DHR cleanup flow.

Technical Reference 302-645, Decay Heat Closed Cycle Cooling Water, Rev. 37.

Open Exam Reference None.

Learning Objective IV.A.11.26

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because DH-V-1 closure would stop dropline flow from the RCS. With stagnant flow conditions at the temperature sensor, indicated suction temperature could not increase under these conditions.

Distracter is plausible because core exit thermocouple temperatures would rise when DHR flow is stopped by closing DH-V-1.

B CORRECT answer. Loss of 1R 480V Bus results in loss of power to DR-P-1A. This presents loss of river water cooling to the DH Service Heat Exchanger, and DHR and RCS temperatrures would rise due to loss of the (river water) heat sink.

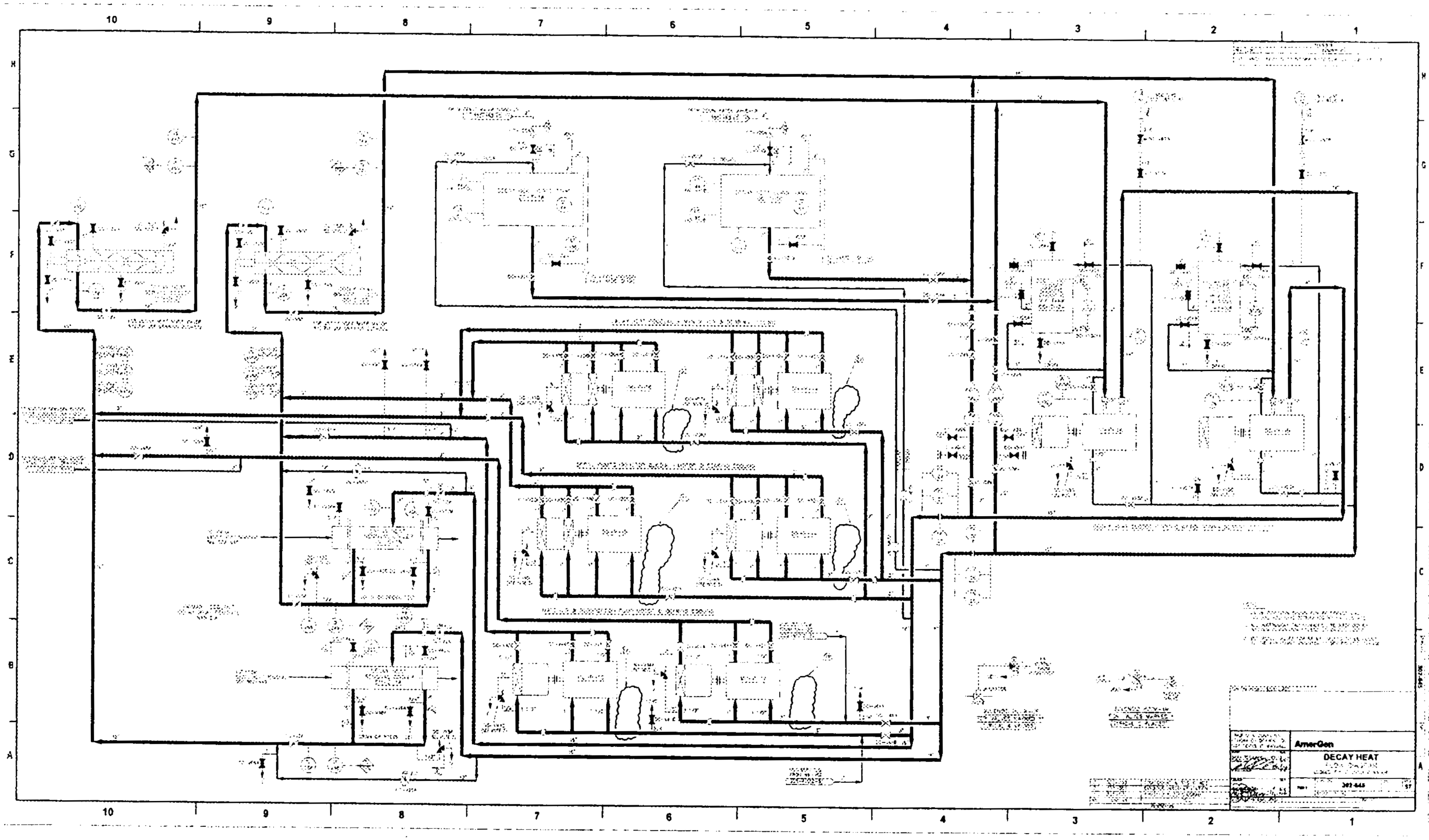
C INCORRECT answer because this operation would not produce noticeable DHR flow changes to produce any significant temperature change.

Distracter is plausible because the increased flow due to venting, if allowed to continue long enough could theoretically result in a DHR suction temperature increase.

D INCORRECT answer because elimination of this flow path would not produce noticeable DHR flow and temperature changes.

Distracter is plausible because it does present a (small) change in DHR dropline flow.

Comments None.


AmerGen
DECAY HEAT
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Form ES-401-5 **Written Exam Question Worksheet** Q # 046

Examination Outline Cross-Reference Tier # 1

Evolution/System 026 Loss of Component Cooling Water Group # 1

K/A # AA2.01 Page # 4.2-19 RO/SRO Importance Rating 2.9 3.5

Measurement Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: Location of a leak in the CCWS.

10CFR55.41(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor is operating at 100% power with ICS in full automatic.
- Intermediate Closed Cooling pump IC-P-1A operating.
- Both Intermediate Service Coolers IC-C-1A and IC-C-1B in service.
- Both Letdown Coolers MU-C-1A and MU-C-1B in service.
- Nuclear Services River Water pumps NR-P-1A and NR-P-1C operating.
- Intermediate Closed Cooling radiation monitor RM-L-9 indication at 1E3 CPM due to an earlier operational problem.
- RM-L-7 plant effluent monitor is in service.

Event:
- MAP C-3-2, IC Surge Tank Level Hi/Lo actuated.

Current conditions:
- IC-T-1 level is 7 inches, LOWERING at 1 inch per minute.
- There have been NO CHANGES in any radiation monitor readings.

Based on these conditions, identify the ONE selection below that describes the location of the leak.

A. Inside one of the Letdown Coolers.

B. CRD cooling outlet pipe inside the RB.

C. Intermediate Service Cooler (IC-C-1A/B).

D. Inside one of the RCP Thermal Barrier Heat Exchangers.

Technical Reference OP-TM-MAP-C0302, Step 4.2.2, Page 3, Rev. 0.
1202-17, Loss of Intermediate Closed Cooling System, Rev 20.

Open Exam Reference None.

Learning Objective IV.B.03.12

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because leakage at this component would RAISE IC-T-1 level.

Distracter is plausible because both coolers are in service.

B CORRECT answer. Refer to 1202-17, Loss of Intermediate Closed Cooling System.

C INCORRECT answer because RM-L-7 indication would rise due to high activity in the ICCW system (included in the question stem).

Distracter is plausible because leakage at this location would be out of the system, and would result in a reduction in IC-T-1 level.

D INCORRECT answer because leakage at this location would RAISE IC-T-1 level.

Distracter is plausible because Intermediate Closed Cooling cools the RCP thermal barrier heat exhangers.

Comments None.

4.2.2 **If** ESAS HPI Channel A/B has **not** actuated,
then PERFORM the following:

1. **MAINTAIN** surge tank level IAW OP-TM-541-463, IC-T-1 Level Control.
2. **CHECK** system piping/components to determine leak location.
3. **If** surge tank level cannot be maintained > 8",
then INITIATE 1202-17, Loss of ICCW.

AmerGen	TMI - Unit 1 Emergency Procedure	Number **1202-17**
Title **Loss of Intermediate Cooling System**		Revision No. **20**
Applicability/Scope TMI Division	**USAGE LEVEL** **1**	Effective Date **07/07/03**

This document is within QA plan scope	X	Yes		No
50.59 Applicable	X	Yes		No

List of Effective Pages

Page	Revision	Page	Revision	Page	Revision	Page	Revision
1	**20**						
2	**20**						
3	**20**						
4	**20**						

1.0 **SYMPTOMS**

1. I.C. Pump Disch Press Lo, Alarm, 70 psig. (C-2-4)
2. I.C. System Flow Lo, Alarm, 550 GPM. (C-2-2)
3. I.C. CRD Clg Flow Lo, Alarm, 100 GPM. (C-1-2)
4. I.C. CRD Clg Outlet Temp. Hi, Alarm, 160°F. (C-1-3)
5. I.C. Surge Tank Level Hi/Lo, Alarm Hi 24"; Lo 8" (C-3-2) IC-LS-802 or 803
6. I.C. Surge Tank Level "A" Hi/Lo Alarm, Hi 24"; Lo 12"; Lo-2 8". (Computer Pt A0451)
7. I.C. Surge Tank Level "B" Lo, Alarm 8". (Computer Pt A0452)
8. I.C. Cooler Outlet Temp. Hi, Alarm, 120°F. (C-2-3)
9. CRD Stator Temp. Hi, 160°F. Computer Point Area 10 Groups (31-38)
10. I.C. R.C. Pump Cooling Outlet Temp. Hi. (Computer Points A0490, A0491, A0492 and A0493) Setpoint 140°F.

2.0 **IMMEDIATE ACTIONS**

2.1 Automatic Actions

- Standby IC Pump starts (ICCW flow less than 550 GPM)
- MU-V-1A/1B closes (CRD Coolant Outlet Hi Temp. greater than 160°F)

2.2 Manual Action

2.2.1 IF low flow exists, THEN **PERFORM** the following:

2.2.1.1 **VERIFY** OR **START** the standby IC pump.

2.2.1.2 **MONITOR** Surge Tank Level.

2.2.1.3 **FILL** Surge Tank Level as necessary to maintain a normal indicated level of 18.5".

3.0 **FOLLOW-UP ACTION**

Objective:

The objective of this procedure is to reestablish cooling flow to CRDM's and RCP seals.

> **NOTE**
>
> IC-P-1A/1B ARE LOCKED OUT ON 27/86 LOCKED OUT. E.S. SIGNAL AND 27/86 LOCKOUT MUST BE RESET BEFORE PUMPS ARE STARTED.

_____ 3.1 IF neither pump can be started, THEN **VERIFY** reset or reset 27/86 lockout relay on PCR for 1P and 1S buses.

3.2 IF both 1C pumps are **NOT** operating, THEN **PERFORM** the following:

_____ 3.2.1 **VERIFY** seal injection flow is greater than 22 GPM as indicated by MU-42FI on console "CC".

_____ 3.2.2 IF seal injection flow is less than 22 GPM AND Intermediate Closed Cooling Water has been lost, THEN **PERFORM** one of the following:

_____ 3.2.2.1 IF Reactor is not reset, **VERIFY** the RCPs have tripped.

3.2.2.2 IF the RCP's have **NOT** tripped, THEN **PERFORM** the following:

_____ (1) **TRIP** the Reactor.

_____ (2) **TRIP** all RCPs.

_____ (3) **VERIFY** OR **START** EFW.

_____ 3.2.2.3 IF Reactor Coolant Pumps trip, THEN **PERFORM** OP-TM-EOP-001, Reactor Trip.

_____ 3.3 IF standby pump started due to tripping of running pump, THEN **INVESTIGATE** for pump/motor problems on the tripped pump.

_____ 3.4 **LOOK** for leakage AND **ISOLATE** portions of the system as needed to keep flow to the CRDs.

_____ 3.5 **MONITOR** CRD Stator temperatures and No. 1 RCP Seal Outlet Water temperature.

> **NOTE**
>
> 1202-8, CRD Equipment Failure, requires the reactor to be tripped if two or more CRD Stator temperatures exceed 180°F.

_____ 3.6 IF CRD Stator high temperature alarms are received, THEN **REFER** 1202-8, CRD Equipment Failure.

____ 3.7 IF RCP Seal Outlet Water high temperature alarm is received, THEN **REFER** 1203-15, Loss or RC Makeup/Seal Injection.

3.8 IF either of the following conditions occur

- CRD Outlet Temp Hi
- CRD Cooling Flow Low

THEN **PERFORM** the following:

____ 3.8.1 **DETERMINE** filter d/p.

____ 3.8.2 IF filter d/p exceeds 10 psid (max. is 12 psid) THEN **CHANGE** the standby filter in accordance with 1104-8, Intermediate Cooling System.

____ 3.9 IF CRD Flow is low, THEN **VERIFY** IC-V6 is open.

3.10 IF ICCW System temperature is high, THEN **PERFORM** the following:

____ 3.10.1 **VERIFY** NSRW System pressure is normal (30-40 psig) as indicated on NR-PI-217.

____ 3.10.2 **PLACE** the standby IC Cooler in service, if needed, in accordance with 1104-30, Nuclear River Water.

____ 3.10.3 **BACKWASH** the inservice IC Cooler in accordance with 1104-30, Nuclear River Water.

____ 3.11 IF letdown flow was isolated, THEN **RE-ESTABLISH** letdown flow to at least 45 GPM in accordance with OP-TM-211-950, Makeup and Purification System. Otherwise N/A this Step.

____ 3.12 IF letdown **CANNOT** be re-established, THEN **COMMENCE** unit shutdown when Pressurizer level reaches approximately 330" in accordance with 1102-10, Plant Shutdown.

3.13 IF Pressurizer level is greater than 380", THEN **PERFORM** the following:

____ 3.13.1 **TRIP** the Reactor.

____ 3.13.2 **PERFORM** OP-TM-EOP-001, Reactor Trip.

Form ES-401-5 **Written Exam Question Worksheet** Q # 047

Examination Outline Cross-Reference Tier # 1

Evolution/System 029 Anticipated Transient Without Scram (ATWS) Group # 1

K/A # EK1.01 Page # 4.1-9 RO/SRO Importance Rating 2.8 3.1

Measurement Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior.

10CFR55.41(8) Components, capacity, and functions of emergency systems.
10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .8/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Initial plant conditions:
- Reactor operating at 100%.
- ICS in automatic EXCEPT Steam Generator/Reactor Master (in MANUAL).
- DSS Diverse Scram System manual pushbutton is NOT operable.

Event:
- Main Turbine trip and both FW Pumps trip due to Main Condenser air in-leakage.
- EFW actuation due to low level in OTSG 1A and OTSG 1B.
- OTSG pressures are both oscillating, responding to MSSV operation.
- Atmospheric Dump Valve MS-V-4A is wide open.
- Atmospheric Dump Valve MS-V-4B is FAILED closed.
- RCS pressure is 2400 psig.
- Manual reactor trip pushbutton has FAILED to initiate a reactor trip.

Based on these conditions identify the ONE selection below that describes:
(1) Most limiting parameter with regards to Primary-to-Secondary heat removal CAPACITY.
(2) Controlling procedure actions to be used to expedite reactor power reduction.

A. (1) ADV and MSSV combined steam flow capacity.
(2) Initiate HPI.

B. (1) ADV and MSSV combined steam flow capacity.
(2) Manually reduce SG/Rx Master to 0% output.

C. (1) EFW system flow capacity.
(2) Initiate HPI.

D. (1) EFW system flow capacity.
(2) Manually reduce SG/Rx Master to 0% output.

Technical Reference OP-TM-EOP-001, Reactor Trip, Step 2.2, Page 1, Rev. 5.
Main Steam System and Turbine Bypass Lesson Plan 11.2.01.063, Page 4, Rev. 15.

Open Exam Reference None.

Learning Objective IV.C.01.05, V.E.13.05

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because even though MS-V-4A can remove approximately 3.2% power, the MSSVs are capable of removing in excess of 100% power.

Distracter is plausible because Part (2), initiating HPI, is the correct method to be used to reduce reactor power.

B INCORRECT answer because even though MS-V-4A can remove approximately 3.2% power, the MSSVs are capable of removing in excess of 100% power. In addition, reactor power reduction using the ICS is not used in EOP-001, the controlling procedure for the ATWS.

Distracter is plausible because Part (2), using the SG/Rx Master to reduce power, is supported by question stem conditions.

C CORRECT answer. With all 3 EFPs operating, the OTSGs can remove approximately 7% power, which is signifcantly less than the combined steaming capacity of one ADV and the MSSVs.

D INCORRECT answer because it includes the wrong method for reducing reactor power.

Distracter is plausible because Part (1) is correct, and Part (2), using the SG/Rx Master to reduce power, is supported by question stem conditions.

Comments None.

REACTOR TRIP

1.0 ENTRY CONDITIONS

- Any unplanned condition requiring an automatic or manual trip signal.
- A symptom of core cooling upset occurs while shutdown prior to DHR operation.

2.0 IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 2.1 **TRIP** the Reactor by depressing the "Reactor Trip" and "DSS" pushbuttons.	
_____ 2.2 **VERIFY** the reactor is shutdown.	_____1. **If** Main FW is **not** available, **then** _____ **ENSURE** Main Turbine is tripped _____ **ENSURE** EFW is actuated _____2. **MAINTAIN** primary-to-secondary heat transfer. _____3. **When** RCS pressure < 2500 psig, **then INITIATE** HPI. _____4. **When** the reactor is shutdown, **then CONTINUE**
_____ 2.3 **TRIP** the Turbine.	
_____ 2.4 **VERIFY** the turbine stop valves are closed.	_____ **PLACE** EHC-P-1A **and** EHC-P-1B in pull-to-lock. _____ **OPEN** EHC-V-FV1 (TB 305', EHC bypass valve at EHC pump skid).

Content/Skills	Activities/Notes
4. Main Steam to Atmosphere This flow path enables controlled relief/discharge of steam to the atmosphere when the Main Condenser is not available. It also is used to provide additional bypass system relief capacity for transients, even when the Main Condenser is operable. The latter function serves to limit peak pressure following a low power turbine trip. Both MS-V-4A/4B are located on the 295' elevation of the Intermediate Building.	**PPT-8** **Actual MSSV capacity of 13,597,902 lb/hr at 3% accumulation.** **Assumes a 114% power overage in the MSSV sizing where most plants utilize 112%.**
5. EF-U-1 Steam Supply Main Steam from the OTSGS and Auxiliary Steam are used to drive the EF-P-1 turbine, which is located on the 295' elevation of the Intermediate Building.	**PPT-9**
6. Gland Seal Steam Supply Through manual isolation valve MS-V-7, OTSG 1B (steam lead #4, downstream of the MS-V-5B connection) provides steam to the Main Turbine Gland Sealing System (to GS-V-4 and GS-V-5). Supplied by Auxiliary Steam when vacuum conditions are initially established in the Main Condenser, the gland seal steam supply is shifted from Auxiliary Steam to Main Steam at the very end of the Plant Heat-up procedure. Later during Plant Startup the gland seal steam source will be shifted from the Main Steam System to the high-pressure turbine gland seal leak-off as Turbine load is increased (self-sealing system design capabilities).	**PPT-10** **Point out that closure of MS-V-1D AND MS-V-1C would be required to terminate an unisolable leak on the GS supply line, since steam could flow from steam lead #3 through the MS-V-5B cross connect pipe into steam lead #4 to feed the leak. This would also be true for leaks on the MS-V-5A/5B pipe penetrations.**
D. Components	**Interim Summary** **Review system functions** **Review flowpath**
1. MS-V-1A/1B/1C/1D These valves are 24" motor operated angle stop check valves, located in the overhead on the 322' elevation of the Intermediate Building.	**PPT-11 Obj. 1.3**

Form ES-401-5 **Written Exam Question Worksheet** Q # 048

Examination Outline Cross-Reference Tier # 1

Evolution/System 038 Steam Generator Tube Rupture (SGTR) Group # 1

K/A # EA1.09 Page # 4.1-11 RO/SRO Importance Rating 3.2 3.3

Measurement Ability to operate and monitor the following as they apply to a SGTR: PZR tank level/pressure indicators, gauges, and recorder.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- Forced plant shutdown is in progress due to OTSG tube leakage,

Based on this condition identify the ONE selection below that describes conditions requiring the operator to manually trip the reactor, based on the relationship between Pressurizer level and reactor power.

A. Pressurizer level 140 inches;
Reactor power 30%.

B. Pressurizer level 140 inches;
Reactor power 20%.

C. Pressurizer level 160 inches;
Reactor power 30%.

D. Pressurizer level 160 inches;
Reactor power 20%.

Technical Reference OP-TM-EOP-005, Step 3.3.1.A, Page 1, Rev. 2.

Open Exam Reference None.

Learning Objective V.E.17.02

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer in accordance with OP-TM-EOP-005. The operator is required to manually trip the reactor if Pressurizer level is less than 150 inches with reactor power greater than 25%.

B INCORRECT answer because reactor power is less than 25%.

Distracter is plausible because the operator is required to manually trip the reactor if Pressurizer level is less than 150 inches with reactor power greater than 25%.

C INCORRECT answer because Pressurizer level is greater than 150 inches.

Distracter is plausible because the operator is required to manually trip the reactor if Pressurizer level is less than 150 inches with reactor power greater than 25%.

D INCORRECT answer because Pressurizer level is greater than 150 inches and because reactor power is less than 25%.

Distracter is plausible because the operator is required to manually trip the reactor if Pressurizer level is less than 150 inches with reactor power greater than 25%.

Comments Question links SGTR event to operation and monitoring Pressurizer level, by addressing manual reactor trip criteria in effect (only during SGTR) based on low pressurizer level and reactor power.

Exelon Nuclear

OTSG TUBE LEAKAGE

1.0 ENTRY CONDITIONS - OTSG tube leakage greater than 1 gpm **and** DHR not in operation.

2.0 IMMEDIATE ACTIONS - None

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ TIME	
_____ 3.1 **NOTIFY** RAC to begin offsite dose assessment.	
_____ 3.2 **INITIATE** Guide 9, "RCS Inventory Control.	
_____ 3.3 **VERIFY** the reactor is critical.	_____ **GO TO** Step 3.4.

NOTE

Use 1102-4, "Power Operation" to control plant power reduction and supplement those actions with the actions in this procedure. The rate of power reductions should be selected to maintain control and avoid lifting MSSVs.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 3.3.1 **INITIATE** a plant shutdown IAW 1102-4, "Power Operation".	
☐ A. **IAAT** pressurizer level < 150 inches, **and** reactor power > 25%, **then** _____ **INITIATE** HPI _____ **TRIP** the reactor. _____ **GO TO** OP-TM-EOP-001.	
☐ B. **IAAT** the turbine trips **and** reactor power > 15%, **then** _____ **TRIP** the reactor. _____ **GO TO** OP-TM-EOP-001.	

Form ES-401-5 **Written Exam Question Worksheet** Q # 049

Examination Outline Cross-Reference Tier # 1

Evolution/System 040 Steam Line Rupture Group # 1

K/A # 2.1.2 Page # 2-1 RO/SRO Importance Rating 3.0 4.0

Measurement Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Initial plant conditions:
- Plant startup in progress with reactor at 15% power.

Sequence of events:
- Reactor trip due to low RCS pressure.
- EOP-001 Reactor trip immediate actions completed.
- Operators diagnosed excessive primary-to-secondary heat transfer at OTSG 1B.
  - OTSG 1B Phase 1 isolation performed.
- Automatic EFW actuation.

Current conditions:
- OTSG 1B level is 3 inches, steady.
- OTSG 1B pressure is 500 psig, steady.

Based on these conditions identify the ONE selection below that represents a steam leak which requires the operator to continue with Phase 2 isolation of OTSG 1B.

A. FW Pipe rupture at Main FW nozzle header.

B. Stuck open Main Steam safety valve MS-V-17B.

C. FW-P-1B steam supply line rupture upstream of MS-V-5B.

D. Failed open turbine bypass valves MS-V-3D, MS-V-3E and MS-V-3F.

Technical Reference OP-TM-EOP-010 Rule 3, Excessive Heat Transfer, Page 6, Rev. 3.

Open Exam Reference None.

Learning Objective V.E.15.02

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer. OP-TM-EOP-003 Rule 3 requires Phase 2 Isolation for steam leaks inside the RB - whether or not OTSG parameters stabilize.

B INCORRECT answer because this leak is not inside the Intermediate Building or the Reactor Building, and OTSG pressure/level have stabilized - therefore Rule 3 does not require Phase 2 Isolation. Even though the MSSVs are located inside the Intermediate Building the leak is to the outside through the discharge pipes that penetrate the roof.

Distracter is plausible because it presents a steam leak that can produce excessive primary-to-secondary heat transfer symptoms, and the MSSVs are located inside the Intermediate Building. This distracter merits additional plausibility since Phase 2 Isolation would be required for this leak if OTSG level and pressure did not stabilize as described in the question stem.

C INCORRECT answer because this leak is in the Turbine Building, and it would be isolated during Phase 1 operations.

Distracter is plausible because it presents a steam leak and location with potential for significant safety impact and personnel injury.

D INCORRECT answer because this leak is not inside the Intermediate Building or the Reactor Building. Also, since OTSG pressure/level have stabilized Rule 3 does not require Phase 2 Isolation. This leak is actually isolated in Phase 1 Isolation operations.

Distracter is plausible because it presents a steam leak that will produce excessive primary-to-secondary heat transfer symptoms.

Comments None.

XHT

3

Rule 3
Excessive Heat Transfer

A. **IAAT** MFW flow is excessive **and** Reactor is shutdown, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** OTSG level < 97.5%.	**TRIP** both Main FW Pumps.
2. **VERIFY** FW Flow is controlled by ICS.	**ADJUST** MFW Pumps **and** FW regulating valves to control OTSG level and valve DP.

B. **IAAT** Primary to Secondary Heat Transfer is excessive **and** Reactor is shutdown, **then**:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **PERFORM** Phase 1 Isolation of the affected OTSG(s).	
2. **VERIFY** OTSG level and pressure stabilizes.	**PERFORM** Phase 2 Isolation of the affected OTSG(s).
3. **VERIFY** steam leak is **not** in RB **or** Intermediate Building.	**PERFORM** Phase 2 Isolation of the affected OTSG(s).
4. **INITIATE** Guide 12, "RCS stabilization following OTSG Isolation".	

Form ES-401-5 **Written Exam Question Worksheet** Q # 050

Examination Outline Cross-Reference Tier # 1

Evolution/System 054 Loss of Main Feedwater (MFW) Group # 1

K/A # AA1.01 Page # 4.2-35 RO/SRO Importance Rating 4.5 4.4

Measurement Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): AFW controls, including the use of alternate AFW sources.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor/Turbine tripped due to low vacuum in the Main Condenser.
- Both FW Pumps tripped.
- All 3 EFW Pumps operating.
- OTSG levels at automatic level control setpoints.
- Atmospheric Dump Valves MS-V-4A/B controlling OTSG pressures at 1010 psig.

Event:
- MAP J-3-3, Condensate Stor Tk A Lvl Lo-Lo, actuates.
- MAP J-3-4, Condensate Stor Tk B Lvl Lo-Lo, actuates.

Based on these conditions identify the ONE selection below that describes:
(1) Next source of water to be used for EFW suction.
(2) When the water supplies are required to be changed.

NOTE:
For purposes of this question, immediately means within the next 2-3 minutes.

A. (1) Main Condenser Hotwell (CO-C-1).
(2) Immediately.

B. (1) Demineralized Water Storage Tank (DW-T-2).
(2) Immediately.

C. (1) Main Condenser Hotwell (CO-C-1).
(2) When Condensate Storage Tank levels are reduced to 2 feet.

D. (1) Demineralized Water Storage Tank (DW-T-2).
(2) When Condensate Storage Tank levels are reduced to 2 feet.

Technical Reference OP-TM-MAP-J0303, Cndensate Stor Tk A Lvl Lo-Lo, Rev. 0.
OP-TM-MAP-J0304, Cndensate Stor Tk B Lvl Lo-Lo, Rev. 0.
OP-TM-EOP-010 Guide 17, Alternate Inventory for Emergency Feedwater, Step 1, Page 37, Rev. 3.

Open Exam Reference None.

Learning Objective IV.C.05.02

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because there is 67,000 gallons remaining until the 2 foot level is reached. Both ARPs referenced direct the operator to GO TO EOP-010 (Guide 17) which applies when CST level reaches 5 feet.

Distracter is plausible because it includes the correct source of water to be used, and it is customary to perform MAP actions as soon as possible.

B INCORRECT answer because there is 67,000 gallons remaining until the 2 foot level is reached. Both ARPs referenced direct the operator to GO TO EOP-010 (Guide 17) which applies when CST level reaches 5 feet. Also, DW-T-2 is not the next source of water to be used.

Distracter is plausible because it includes a bonafide source of water that can be used, and it is customary to perform MAP actions as soon as possible.

C CORRECT answer. Refer to OP-TM-MAP-J0303, OP-TM-MAP-J0304, and EOP-010 Guide 17.

D INCORRECT answer because the source of water identified is incorrect.

Distracter is plausible because it includes a bonafide source of water that is available (used later), and it correctly states when the suction source transition is to be completed.

Comments None.

Level 2 – Reference Use

1.0 SETPOINTS

- CO-LI-1060 < 5'

2.0 CAUSES

- Leak in the Condensate Tanks or Secondary Systems
- Secondary inventory being depleted by steaming to atmosphere

3.0 AUTOMATIC ACTIONS - None

4.0 MANUAL ACTIONS REQUIRED

- **OBSERVE** the following:

CO-T-1A	CO-T-1B	CO-C-1
CO-LI-1061 (CC)	CO-LI-1063 (CC)	CO-LT-42 (CL)
CO-LI-1060 (PLF)	CO-LI-1062 (PLF)	N/A

NOTE: This alarm means that there is a 78 min. supply to the suction of the Emergency Feedwater pumps, before they lose suction.

- **If** EFW is in operation,
  then GO TO OP-TM-EOP-010 for alternative condensate supply.

NOTE: This action is to prevent a system failure from potentially drawing down both tanks. If level control is established, these valves may be reopened as necessary.

- **If** cause of level drop is **not** known,
  or a major system leak is suspected,
  then CLOSE the following valves:
  - CO-V-14A (CC)
  - CO-V-14B (CC)
  - CO-V-111A (CC)
  - CO-V-111B (CC)
- **INITIATE** application of appropriate administrative control for components **not** in the required mode position (Clearance / EST).

Level 2 – Reference Use

1.0 SETPOINTS

– CO-LI-1063 < 5'

2.0 CAUSES

– Leak in the Condensate Tanks or Secondary Systems

– Secondary inventory being depleted by steaming to atmosphere

3.0 AUTOMATIC ACTIONS - None

4.0 MANUAL ACTIONS REQUIRED

– **OBSERVE** the following:

CO-T-1B	CO-T-1A	CO-C-1
CO-LI-1063 (CC)	CO-LI-1061 (CC)	CO-LT-42 (CL)
CO-LI-1062 (PLF)	CO-LI-1060 (PLF)	N/A

NOTE: This alarm means that there is a 78 min. supply to the suction of the Emergency Feedwater pumps, before they lose suction.

– **If** EFW is in operation,
then GO TO OP-TM-EOP-010 for alternative condensate supply.

NOTE: This action is to prevent a system failure from potentially drawing down both tanks. If level control is established, these valves may be reopened as necessary.

– **If** cause of level drop is **not** known,
or a major system leak is suspected,
then CLOSE the following valves:

  – CO-V-14A (CC)
  – CO-V-14B (CC)
  – CO-V-111A (CC)
  – CO-V-111B (CC)

– **INITIATE** application of appropriate administrative control for components **not** in the required mode position (Clearance / EST).

Guide 17
Alternate inventory for Emergency Feedwater
(Page 1 of 3)

_____ 1. **When** condensate storage tank level reaches five (5) feet, **then**:

NOTE

At 5 ft on CST there is 67,000 gallons remaining
(≈ 20 minutes of usage until a level of 2 feet is reached).

_____ 1.1 **ENSURE** the spectacle flange EFV4/5 has been swapped or action is in progress.

_____ 1.2 **PERFORM** OP-TM-331-151 “BREAKING VACUUM”

1.3 **UNLOCK and CLOSE** the following breakers for EF-V-4 and 5:

_____ EF-V-4 (1C ES Valves MCC Unit 7A)

_____ EF-V-5 (1C ES Valves MCC Unit 7B)

_____ 2. **When** Condensate Storage Tank level reaches approximately two (2) feet, **then**:

_____ 2.1 **VERIFY** condenser is within 1 “Hg of atmospheric pressure.

NOTE

The "EMERG" positions of these valves is as follows:

CO-V-6 - Close – prevents air from entering the pump suction line.
CO-V-7 - Close – prevents air from entering the pump suction line.
CO-V-8 - Open – permits the pump to take suction directly from the Hotwell.

_____ 2.2 **PRESS** the "EMERG" pushbutton for the “Condensate Makeup and Dump Mode Selector”.

_____ 2.3 **ENSURE** open CO-V-12 **and** CO-V-13.

Form ES-401-5 **Written Exam Question Worksheet** Q # 051

Examination Outline Cross-Reference Tier # 1

Evolution/System 055 Loss of Offsite and Onsite Power, Station Blackout Group # 1

K/A # EA2.02 Page # 4.1-15 RO/SRO Importance Rating 4.4 4.6

Measurement Ability to determine or interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling.

10CFR55.41(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor tripped due to loss of offsite power (LOOP).
- Emergency Generators EG-Y-1A and EG-Y-1B failed.
- EF-V-30A/B/C/D are full open, and EFW flow is balanced between the two OTSGs.
- Parameter

Parameter	Value	Trend
RCS Pressure	1985 psig	Lowering 20 psi/minute
OTSG Pressures	1010 psig	Lowering 10 psi/minute
OTSG 1A Level (Operate Range)	10%	Rising 1% per minute
OTSG 1B Level (Operate Range)	8%	Rising 1% per minute
Core Outlet Temperature*	603 Degrees	Steady
Loop A/B T-Hot	595 Degrees	Steady
Loop A/B T-Cold	565 Degrees	Lowering 2 degrees/minute

* Average of the five highest incore thermocouples.

Based on these conditions, identify the ONE selection below that describes status of:
(1) RCS natural circulation.
(2) Primary-to-secondary heat transfer (PSHT).

A. (1) Natural circulation is NOT occurring.
(2) PSHT is NOT occurring.

B. (1) Natural circulation is NOT occurring.
(2) PSHT is occurring.

C. (1) Natural circulation is occurring.
(2) PSHT is NOT occurring.

D. (1) Natural circulation is occurring.
(2) PSHT is occurring.

Technical Reference OP-TM-EOP-010 Guide 10, Natural Circulation Cooling, Page 25, Rev. 3.
OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 3.13, Primary-to Secondary Heat Transfer, Page 6, Rev. 10.

Open Exam Reference None.

Learning Objective V.E.10.03

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because (1) natural circulation is occurring IAW EOP-010 Guide 10, and (2) PSHT is occurring IAW OS-24.

Distracter is plausible because:
(1) Natural circulation:
- OTSG levels (50% on the Operating Range) have not been established.
- Incore thermocouple core outlet temperature is greater than 600 degrees.
- One could argue that the OTSGs do not have "water level control" (not defined in OS-24). Since OTSG levels are being raised, both have water level control.

(2) PSHT:
- Requires forced or verified RCS natural circulation. If examinee decides natural circulation is not occurring, he must deduce that PSHT cannot be declared.

B INCORRECT answer because natural circulation is occurring IAW EOP-010 Guide 10.

Distracter is plausible because:
Natural circulation:
- OTSG levels (50% on the Operating Range) have not been established.
- Incore thermocouple core outlet temperature is greater than 600 degrees.
- One could argue that the OTSGs do not have "water level control" (not defined in OS-24) because both are below the 50% level control setpoint for natuaral circulation. Since OTSG levels are being raised, both have water level control.

PSHT:
- Both OTSGs are AVAILABLE as heat sinks IAW OS-24 Section 3.10.

C INCORRECT answer because PSHT is occurring.

Distracter is plausible because:
(1) Natural circulation is occurring IAW EOP-010 Guide 10.
(2) Among other requirements (satisfied in the stem) OS-24 requires at least one OTSG to have water level and pressure control in order to declare PSHT. With both OTSGs below the 50% level control setpoint in the question stem, this distracter is plausible since one could argue neither OTSG has "water level control" (not defined in OS-24). This would not be valid since the stem presents rising level in both OTSGs. Technically OTSG (RCS) heat removal can occur prior to declaring formal PSHT, using the OS-24 definition

D CORRECT answer because:
NATURAL CIRCULATION IS PRESENT:
- RCS Delta-T is less than 50 degrees.
- T-Hot is less than 600 degrees.
- Incore temperature has stabilized, and is tracking T-Hot.
- OTSG heat removal exists as indicated by the presence of (EFW) feed flow.
- RCS SCM is (33 degrees) greater than 25 degrees.

PSHT CAN BE DECLARED:
- Both OTSGs have water level control and pressure control (one is required).
- RCS T-Cold is approximately the same as OTSG Tsat and responds to changes in OTSG pressure.
- Natural circulation is present IAW EOP-010 Guide 10.

Comments None.

Guide 10
Natural Circulation

IAAT all RCPs are off, **then**:

A. Natural circulation can be VERIFIED if ALL of the following are TRUE:

- RCS Delta-T rises and stabilizes at less than 50°F.
- T_H < 600°F.
- Incore thermocouple temperature stabilizes and tracks T_H.
- Cold leg temperatures approach saturation temperature for secondary side pressure.
- OTSG heat removal exists as indicated by: Feed flow indication OR Steam flow indication
- SCM $\geq$ 25°F.

______ TIME Natural Circulation was VERIFIED

B. **MAINTAIN** RCS pressure above the "PREVENT RV HEAD BUBBLE" curve on Figure 1 **and CONTROL** RCS cooldown rate to < 50°F/HR to avoid developing a steam bubble in the Reactor Vessel head.

3.13 PRIMARY-TO-SECONDARY HEAT TRANSFER (PSHT):

PSHT is the removal of sensible heat from the RCS to one or both OTSG(s). PSHT can be confirmed if all of the following conditions exist:

- Either OTSG has water level control and pressure control.
- RCS T_c is approximately the same as secondary T_{sat} and responds to changes in OTSG pressure.
- RCS forced or verified natural circulation is present.

3.14 REACTOR SHUTDOWN:

Heat generation by a self sustaining fission process has been effectively stopped. This reactor condition can be assessed immediately following reactor trip by Power Range Nuclear Instrumentation < 7 % FP. Nuclear Instrumentation may not represent reactor power during Loss of Coolant Accidents.

3.15 SYMPTOM CHECK:

A review of plant conditions to determine if a symptom of a core cooling upset exists. Refer to Attachment D, Symptom Check Guidelines.

3.16 TRICKLE FEED:

Trickle Feed is feeding a OTSG that cannot hold pressure (i.e., unisolable steam leak). RCS temperature and cooldown rate are controlled by feedwater flow instead of OTSG pressure.

3.17 VERIFY:

Observe whether a condition exists. No action is intended.

Exception: ENSURE and VERIFY in Emergency (1202 series) and Abnormal (1203 series) procedures are used interchangeably until they are revised.

Form ES-401-5 **Written Exam Question Worksheet** Q # 052

Examination Outline Cross-Reference Tier # 1

Evolution/System 057 Loss of Vital AC Electrical Instrument Bus Group # 1

K/A # AA2.18 Page # 4.2-42 RO/SRO Importance Rating 3.1 3.1

Measurement Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The indicator, valve, breaker, or damper position which will occur on a loss of power.

10CFR55.41(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- ES Panel PCR Indicator Lights Power Supply Selector Switch in the "BUS A AND B" Position.

Event:
- Reactor trip with ES actuations due RCS LOCA.
- Inverter 1A failure results in de-energizing 120V AC Vital Bus 1A.

Based on these conditions identify the ONE selection below that describes the impact on control room position indication as the operator attempts to verify LPI valve DH-V-4A travels to the full open position.

A. Blue, Amber lamps are operable.
Red, Green Lamps are operable.

B. Blue, Amber lamps are operable.
Red, Green Lamps are NOT operable.

C. Blue, Amber lamps are NOT operable.
Red, Green Lamps are operable.

D. Blue, Amber lamps are NOT operable.
Red, Green Lamps are NOT operable.

Technical Reference 208-433, DH-V-4A, Rev 12.
209-502, ES Actuation a ES Panel Indication, Rev 8.
209-489, ES Actuation - ES Panel Indication, Rev. 4.
209-488, ES Actuation - ES Panel Indication, Rev. 10.
209-487, ES Actuation - ES Panel Indication, Rev. 9.

Open Exam Reference None.

Learning Objective IV.E.24.27

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because under these conditions the (ES panel) blue/amber lights are rendered inoperable by the loss of VBA.

Distracter is plausible because the control console red/green lamps (powered by the control power transformer at the MOV breaker) remain operable, and one might think that both indication systems are powered from one common power supply. In addition, with the power supply selector switch in the Bus A and B position, one might think there is an automatic power transfer scheme.

B INCORRECT answer because the ES panel blue/amber lights are rendered inoperable by loss of VBA, and the control console red/green lamps remain operable since they are powered by the control power transformer at the MOV breaker.

Distracter is plausible because the ES panel blue/amber lamps are rendered inoperable by loss of VBA. In addition, with the power supply selector switch in the Bus A and B position, one might think there is an automatic power transfer scheme.

C CORRECT answer.

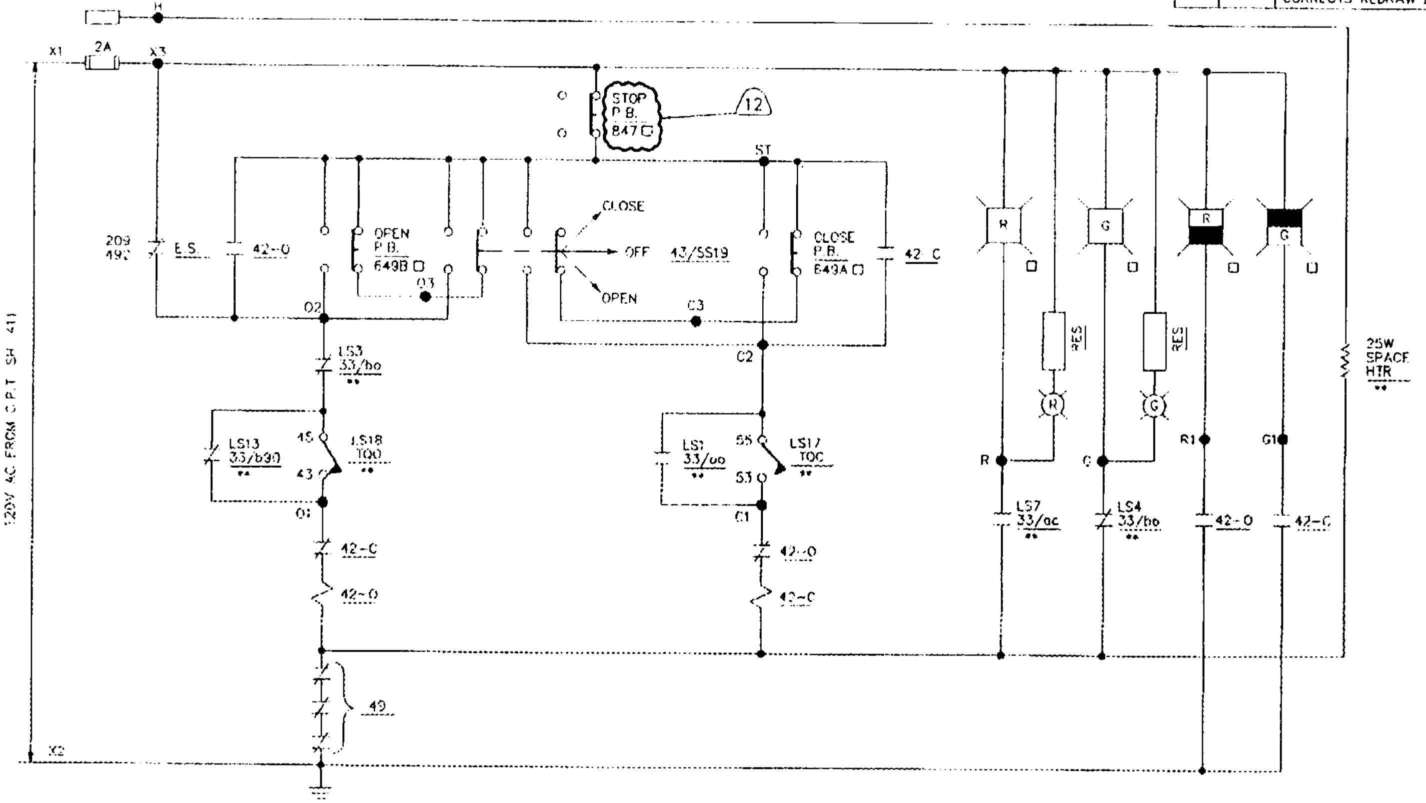
D INCORRECT answer because the power source for the control console red/green lamps (control power transformer at the MOV breaker) is NOT rendered inoperable by loss of VBA.

Distracter is plausible because one might think that both indication systems are powered from one common power supply (VBA).

Comments None.

REVISIONS		
REV	ZONE	DESCRIPTION
12		ADMINISTRATIVE CHANGE – REVISION 11 OF THIS DRAWING WAS RE-DRAWN IN AUTO CAD. WHEN REDRAWN, STOP P.B. WAS SHOWN AS OPEN. REV 12 CORRECTS REDRAW ERROR TO SHOW P.B. AS STOP. ALSO, REMOVED NSR STAMP

VALVE SHOWN IN FULL CLOSED POSITION
LIMITORQUE VALVE OPERATOR

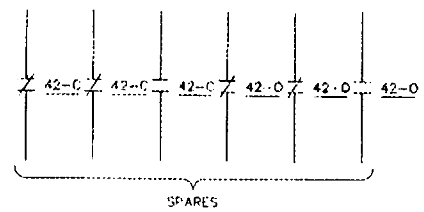


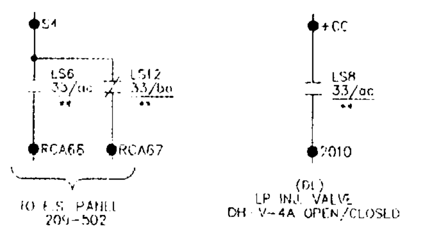
NOTES

1. THE SET POINTS OF LS9 THRU LS12 & LS13 THRU LS16 ARE ADJUSTABLE AT ANY PERCENTAGE OF VALVE POSITION
2. CF-CONSOLE SECT. CC
3. OPEN P.B. 649B TAG NO. PBO/DHV4A
   CLOSE P.B. 649A TAG NO. PBC/DHV4A
   STOP P.B. 847 TAG NO. PBS/DHV4A

REFERENCE DWGS.

INDEX: 208-401
LEGEND: 208-001
SS DEVELOPMENT: 208-413





LIMITORQUE VALVE OPERATOR
VALVE SHOWN IN FULL CLOSED POSITION

LIMIT SWITCH CONTACT DEVELOPMENT

ROTOR	CONT	VALVE POSITION (FULL CLOSED – FULL OPEN)	DESIGN	CONTACT FUNCTION
1	1		33/oo	CLOSE BYPASS
	2		33/oo	SPARE
	3		33/bo	OPEN LIMIT
	4		33/bo	CLOSE IND LT
2	5		33/bc	SPARE
	6		33/oc	OPEN ES IND
	7		33/oc	OPEN IND LT
	8		33/oc	CMPT POINT
3	9		33/oo	SPARE
	10		33/oo	SPARE
	11		33/bo	SPARE
	12		33/bo	CLOSE ES IND
4	13		33/b90	OPEN BYPASS
	14		33/b90	SPARE
	15		33/a90	SPARE
	16		33/a90	SPARE

LS17/TOSW-CLOSING TORQUE SWITCH
LS18/TOSW-OPENING TORQUE SWITCH

Cad File Name : 208433_S1_R12

12

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

AmerGen

ELECTRICAL ELEMENTARY DIAG
480V CONT. CTR. 1A-ESV UNIT 1C
LP INJECTION VALVE DH-V-4A

DRAWN	DATE
T.V. GARBINI	04/24/02
CHECKED	04/29/02
DESIGN MANAGER	4/30/02
ENGINEER	DATE
ENGINEERING MANAGER	DATE

TMI-1 | DWG NO 208-433 | SH 1 | REV 12 | SCALE:

INTERFACING CONCURRENCE | BA NO

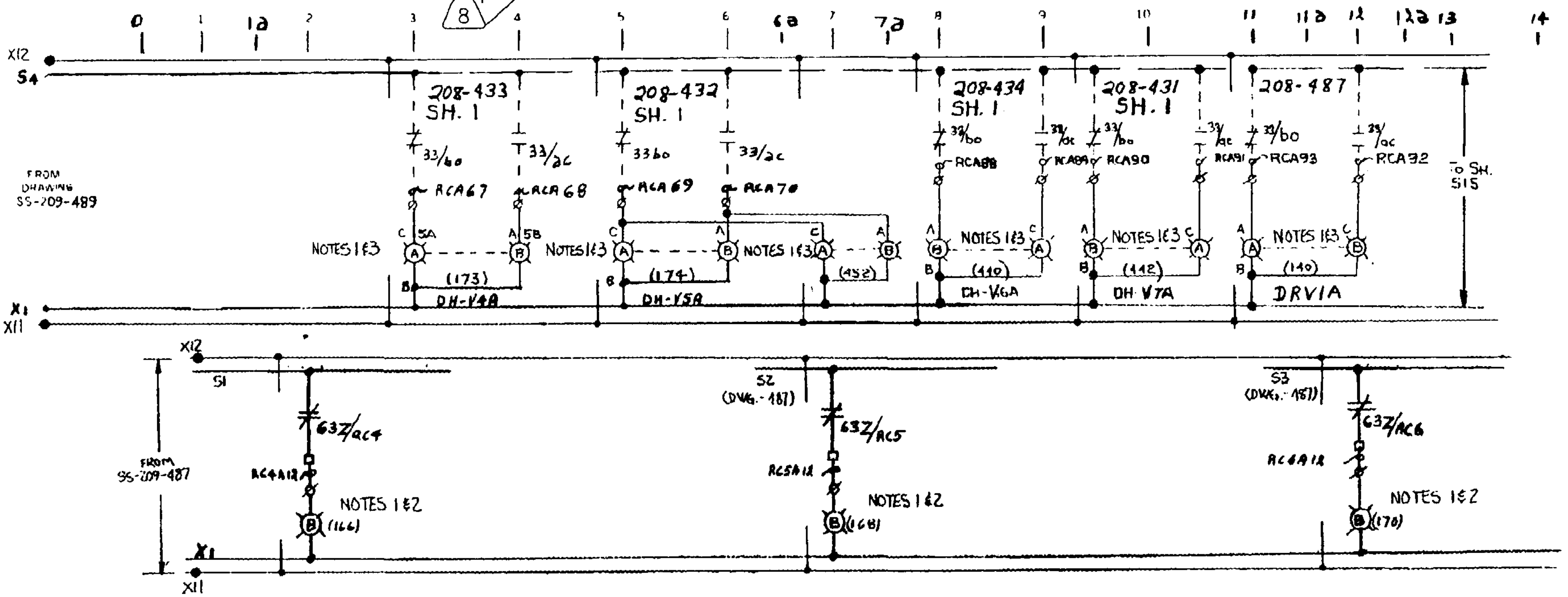
NO	DWG. NO	TITLE
	REFERENCE	

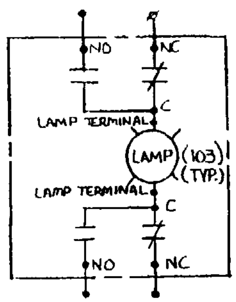
DWG. NO. 208-433 SH 1

METROPOLITAN EDISON COMPANY		MADE CAS 7/69	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA		
THREE MILE ISLAND NUCLEAR STATION UNIT 1		CHK'D JWH 6/70			
		SQ CF GC			
ELECTRICAL ELEM. WIRING DIAGRAM		CF DFN JWH	4192	SS-209-502	8
		ENG	WORK ORDER	SIZE DRAWING	REV
ENGINEERED SAFEGUARD		REV CH APP DATE			

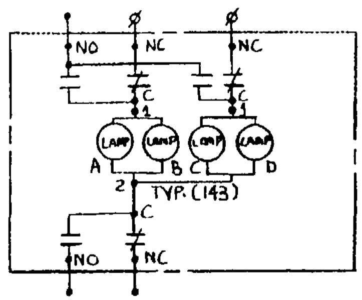
E.S. ACTUATION A E.S. PANEL INDICATION. (PNL.PCR)

LOW PRESSURE INJ. CHANNEL RC4A, RC5A, RC6A - ACTUATION





DETAIL 1



DETAIL 2

NOTES:

1. PUSH-TO-TEST INDICATING LIGHT UNIT.
2. FOR COMPLETE INTERNAL WIRING, SEE DETAIL 1
3. FOR COMPLETE INTERNAL WIRING, SEE DETAIL 2.

CAD FILE: 209502_R8
THIS IS A COMPUTER GENERATED DRAWING.
DO NOT REVISE IT MANUALLY.

AmerGen

REV	DRAFT	CHECK	APPROVED	DATE	APP
8	REVISED TO INCORPORATE (1) FCR 02-00462, AND (2) ADMINISTRATIVE CHANGE				
	TVG	RW		8/28/02	

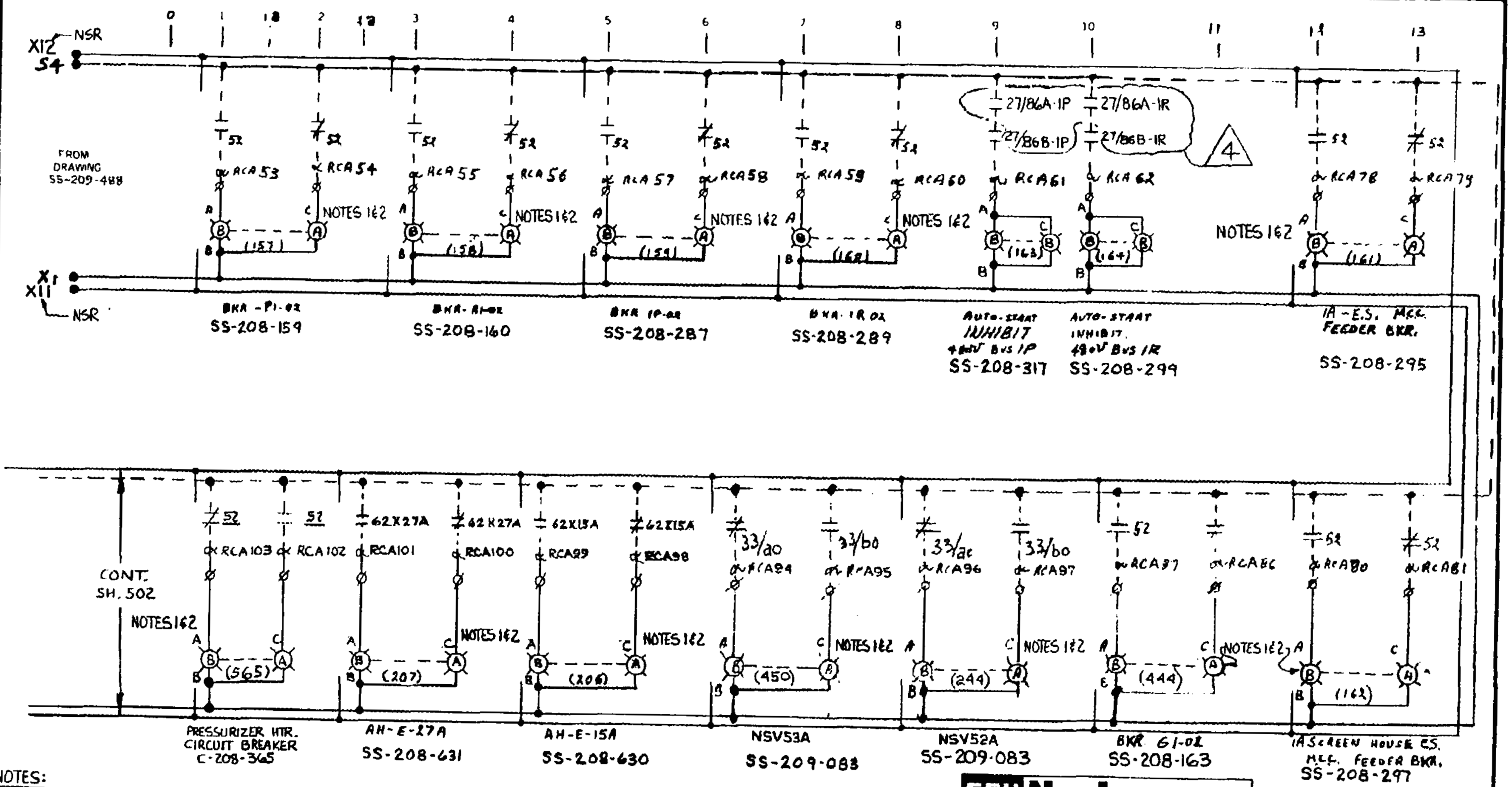
REVISION

NUCLEAR SAFETY RELATED

E.S. ACTUATION - E.S PANEL INDICATION (PNL. PCR)

H.P. INJECTION & LOADING SEQUENCE. - ACTUATION & BYPASS.

METROPOLITAN EDISON COMPANY	MADE GRM 11-70	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA	
THREE MILE ISLAND NUCLEAR STATION UNIT 1	CHK'D VHW 8/70		
ELECTRICAL ELEM WIRING DIAGRAM	SQ CF GC	4192 WORK ORDER	SS-209-489 SIZE DRAWING / 4 REV
ENGINEERED SAFEGUARD	CF DFN JWH / ENG / REV CH APP DATE	2-DP-	



NOTES:

1. PUSH-TO-TEST INDICATING LIGHT UNIT
2. FOR COMPLETE INTERNAL WIRING SEE DETAIL 2, GAI DWG SS-209-502

GPU Nuclear

REV	DESCRIPTION	DRAFT	CHECK	APPROVED	DATE	APP
4	REVISED TO INCORPORATE ECD-C204262	SGM	TVG		2-8-97	

REVISION

REV	DESCRIPTION	MADE	CHKD	PDS	APP	DATE
3	ADDED SAFETY CLASS., REV TO IND. AS BUILT CONDS. PER GPU FCN-C004059, C004971 & GAI CN-0916. INTERFACE: CCX	RJW	RD	RPS	JHS	1-4-83

E.S. ACTUATION "A" - E.S. PANEL INDICATION (PNL. PCR)

H.P. INJECTION & LOADING SEQUENCE - ACTUATION

NOTES:

1. No. Between Brackets Are Panel PCR Items No
2. Push-To-Test Indicating Light Unit.
3. For Complete Internal Wiring See Detail 1, SS-209-502
4. For Complete Internal Wiring See Detail 2, SS-209-502

GPU Nuclear

REV	DRAFT	CHECK	APPROVED	DATE	APP
10	REVISED TO INCORPORATE ECD C204262				
	SGM	TVG	[illegible]	[illegible]	

REVISION

METROPOLITAN EDISON COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT 1

ELECTRICAL ELEM. WIRING DIAGRAM

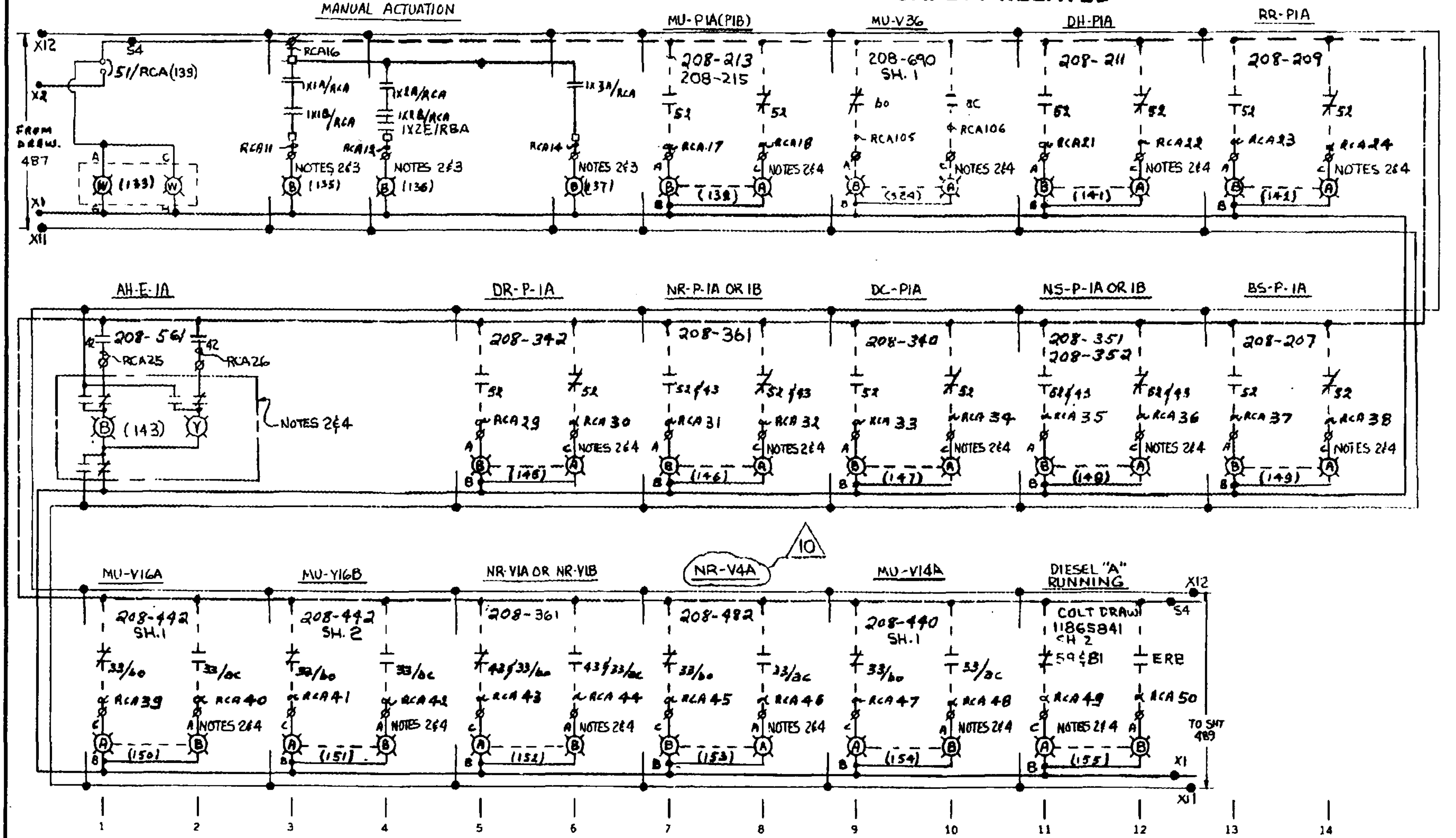
ENGINEERED SAFEGUARD

MADE GRM 11-70
CHK'D VHN 8/70
SQ. CK. GC
CF. DFN. JWH
ENG. [illegible]
REV CH APP. DATE

GILBERT ASSOCIATES, INC.
ENGINEERS AND CONSULTANTS
READING, PENNA.

WORK ORDER	SIZE	DRAWING	REV
4192		SS-209-488	10

NUCLEAR SAFETY RELATED



E.S. ACTUATION "A" - E.S. PANEL INDICATION (PNL PCR)

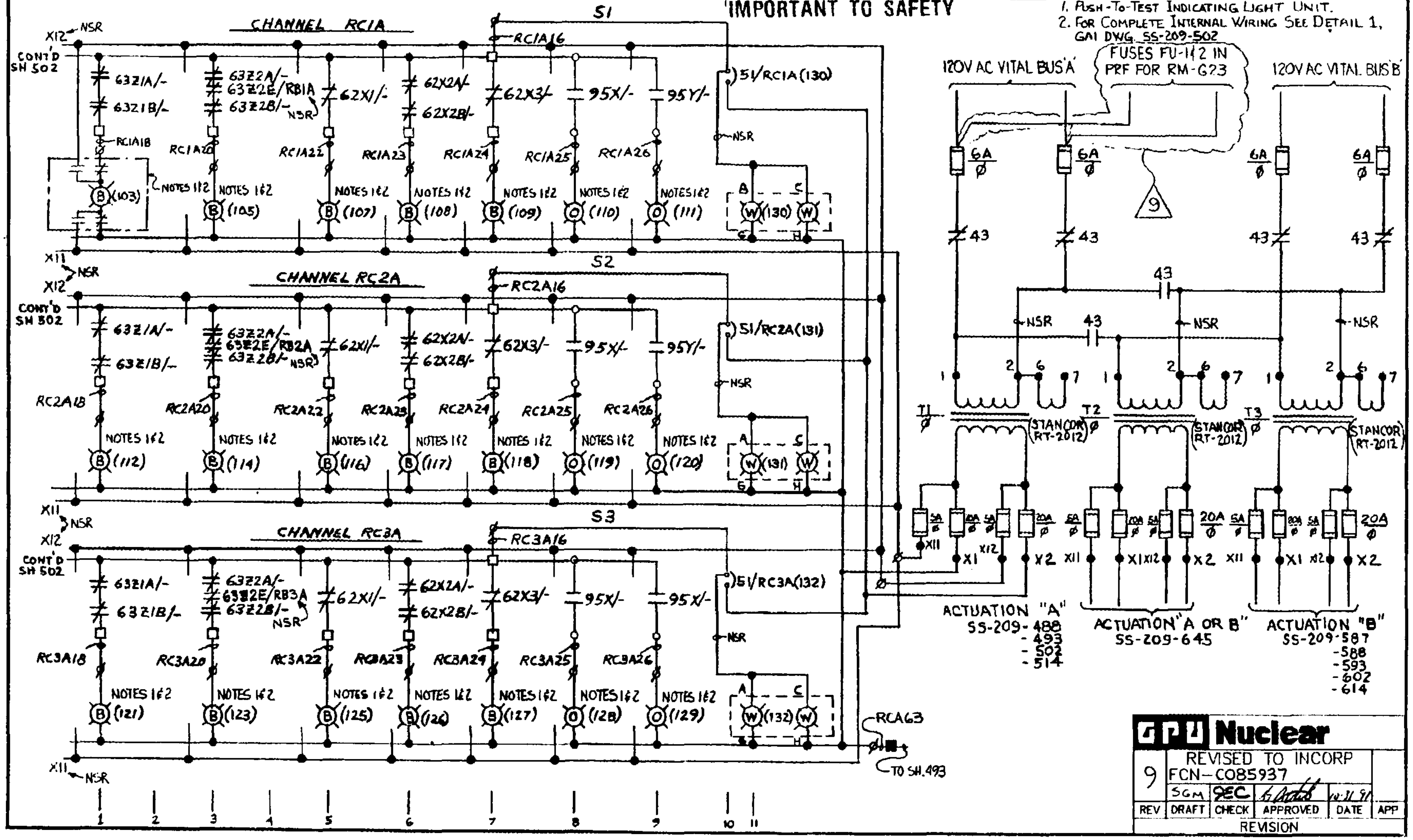
H.R INJECT. & LOADING SEQUENCE CHANNEL RC1A, RC2A, RC3A

METROPOLITAN EDISON COMPANY	MADE GRM 7-70	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA. AND NEW YORK, N.Y.
THREE MILE ISLAND NUCLEAR STATION UNIT 1	CHK'D HN 12/70	
ELECTRICAL ELEMENTARY DIAGRAM	EQ. CF. GC	WORK ORDER 4192 / SIZE DRAWING SS-209-487 / REV. 9
ENGINEERED SAFEGUARD	CP. DEM. JWH	

NUCLEAR SAFETY RELATED
IMPORTANT TO SAFETY

NOTES:

1. PUSH-TO-TEST INDICATING LIGHT UNIT.
2. FOR COMPLETE INTERNAL WIRING SEE DETAIL 1, GAI DWG. SS-209-502



GPU Nuclear

REV	DRAFT	CHECK	APPROVED	DATE	APP
9	SGM	SEC		10-11-91	

9 REVISED TO INCORP FCN-C085937

REVISION

Form ES-401-5 **Written Exam Question Worksheet** Q # 053

Examination Outline Cross-Reference Tier # 1

Evolution/System 058 Loss of DC Power Group # 1

K/A # AA1.02 Page # 4.2-43 RO/SRO Importance Rating 3.1 3.1

Measurement Ability to operate and / or monitor the following as they apply to the Loss of DC Power: Static inverter dc input breaker, frequency meter, ac output breaker, and ground fault detector.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.

Event:
- Fuse failure at Switch #4 125/250 Volt DC Distribution Panel 1A - DC power to Inverter 1A.

Based on these conditions, identify the ONE selection below that describes the operational impact of this event on Inverter 1A.

A. DC input breaker at the inverter will trip open.
Inverter will FAIL because the inverter AC output breaker will trip open.

B. DC input breaker at the inverter will trip open.
Inverter will continue to operate because the AC output breaker will NOT open.

C. DC input breaker at the inverter will NOT trip open.
Inverter will FAIL because the inverter AC output breaker will trip open.

D. DC input breaker at the inverter will NOT trip open.
Inverter will continue to operate because the AC output breaker will NOT open.

Technical Reference MAP A-3-7, Inverter 1A/1C/1E Trouble, Automatic Action 1, Page 1, Rev. 16.

Open Exam Reference None.

Learning Objective IV.G.10.3

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT ANSWER because the inverter will continue to operate, even with a loss of DC input.

Distracter is plausible because the DC input breaker does trip open on low voltage.

B CORRECT answer.

C INCORRECT ANSWER because the DC input breaker does trip open on low voltage and the inverter will continue to operate, even with a loss of DC input.

Distracter is plausible because the DC voltage is lost when the fuse fails, and it is a plausible misconception that the inverter will fail.

D INCORRECT ANSWER because the inverter will DC input breaker does trip open on low voltage.

Distracter is plausible because the inverter will continue to operate, even with a loss of DC input.

Comments Question addresses expected operation (ability to operate and/or monitor) static inverter dc input breaker, frequency meter, ac output breaker on loss of DC power.

ALARM:

INVERTER 1A/1C/1E TROUBLE

SETPOINTS:

DC Volts Lo 101 VDC
DC Volts Hi 144 VDC
Battery Overcurrent 146-150 amps (assumes inverter is on the battery)
Inverter Frequency < 59 or > 61 cycles per second

CAUSES:

Battery voltage high caused by battery charger trouble
Battery voltage low caused by high battery discharge rate (580 amps for one hour, for example) as indicated by the ammeters in DC Dist. Panel 1A.
Overcurrent as indicated on the ammeter "Battery Input Amps" on the front of the inverter, due to an overload of the inverter or inverter malfunction.
Switch open on the DC Dist. Panel feeding the inverter

Inverter	DC Panel	Switch
1A	1A	4
1C	1A	12
1E	1A	11

Inverter out of sync and internal oscillator off frequency
Frequency detector problem

AUTOMATIC ACTION:

1. The DC input breaker trips on over or undervoltage at the alarm point. The inverters should continue to operate from their AC source, if available.
2. If the AC source is not available, the inverter will trip when the DC input breaker opens. This will cause loss of the affected vital bus, affected RPS channel will trip, and trip of 1 channel of RB isolation and cooling, HPI, and LPI in both ES actuation systems will occur.

OBSERVATION (CONTROL ROOM):

Computer Printout: Example "INVERTER 1A DC VOLTS LOW"

Inverter 1A/1C/1E DC Volts Lo L2550/L2558/L2685
Inverter 1A/1C/1E DC Volts Hi L2551/L2559/L2686
Inverter 1A/1C/1E DC Overcurrent L2552/L2560/L2687
Inverter 1A/1C/1E Freq. Hi/Lo L2760/L2612/L2654

Form ES-401-5 **Written Exam Question Worksheet** Q # 054

Examination Outline Cross-Reference Tier # 1

Evolution/System 065 Loss of Instrument Air Group # 1

K/A # AK3.08 Page # 4.2-50 RO/SRO Importance Rating 3.7 3.9

Measurement Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Actions contained in EOP for loss of instrument air.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .5/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Plant conditions:
- Manual reactor trip from 100% power due to low Instrument Air pressure.
- Operators have completed applicable emergency procedure immediate actions.
- Primary and Secondary air pressure indications are at 5 and 10 psig respectively.
- OTSG levels are both 8 inches, and slowly lowering.
- EFW Pumps EF-P-1, EF-P-2A and EF-P-2B are all operating.
- EFW control valves EF-V-30A-D are all closed even though their controller automatic demand signals are all at 100%.

Based on these conditions, identify the ONE selection below that describes why EF-V-30 A-D are closed.

A. Trip of back-up compressor.

B. Low primary plant air system pressure.

C. Low 2-hour air supply system pressure.

D. Low secondary plant air system pressure.

Technical Reference 1202-36, Loss of Instrument Air, Pages 4 and 9, Rev. 34.

Open Exam Reference None.

Learning Objective IV.D.14.11

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because the EF-V-30 valves are supplied by the 2-hour backup air bottles banks, and they fail closed when local air pressure reduces to less than 15 psig.

Distracter is plausible because of the safety significance of these valves. It is a plausible misconception that they are supplied by the backup air compressors.

B INCORRECT answer because the EF-V-30 valves are supplied by the 2-hour backup air bottles banks, and they fail closed when local air pressure reduces to less than 15 psig.

Distracter is plausible because of the safety significance of these valves. It is a plausible misconception that they are supplied by the primary plant air header system.

C CORRECT answer. The EF-V-30 valves are supplied by the 2-hour backup air bottles banks, and they fail closed when local air pressure reduces to less than 15 psig.

D INCORRECT answer because the EF-V-30 valves are supplied by the 2-hour backup air bottles banks, and they fail closed when local air pressure reduces to less than 15 psig.

Distracter is plausible because these valves are physically located near other components supplied by the secondary plant air header system.

Comments None.

CAUTION

Bypassing of the air drier and filters following starting additional compressors is required to prevent exceeding individual component flowrates. High flowrates without bypassing components could possibly cause blockage due to component failures resulting in total loss of IA worsening the casualty.

_____ 2. Dispatch an AO to the IA-Q-1 drier area to perform the following:

_____ 2.1 Open IA-V-16 (Prefilter bypass)

_____ 2.2 Open IA-V-19 (Drier bypass)

_____ 2.3 Open IA-V-25 (After filter bypass)

_____ 2.4 Verify open or open IA-V-2104A/B.

_____ 3. Verify locally that the backup instrument air compressors (IA-P-2A/B) are operating.

_____ 4. If the backup instrument air compressors IA-P-2A(B) are not operating then attempt to start them.

_____ 5. Close IA-V-1028A and B to prevent backflow of CW into IA system via Amertap ΔP gages.

_____ 6. Monitor the following components at the respective instrument air pressure to ensure backup sources are functioning properly. Take local control as necessary.

Instrument Air Pressure	Valves	Consequence if Fail
6.1 55 psig	MU-V-20, IC-V-3,4,6	RCP Trip, loss of RCP seal cooling and seal injection
6.2 35 psig	MS-V-3's,4's,6 FW-V-16,17's FW-P-1A Speed Control	Primary to Secondary Heat Transfer Upset
6.3 15 psig	EF-V-30's	Primary to Secondary Heat Transfer Upset

_____ 7. Monitor pressure in 2-hour backup air bottle banks on local indicators IA-PI-1011/1012. Evaluate need for recharging.

NOTE

EF-V-30A/B/C/D, MS-V-4A, 4B, 6 and RR-V-6 are supplied by a 2 hour air bottle supply. MS-V-6 is initially supplied from 2 hour Train "A". If Train "A" is unavailable, Train "B" can supply MS-V-6 by opening IA-V-1632.

TABLE 1

Air Operated Valve Number	Local Pressure at which component position becomes unreliable	Air Fail Position
CO-V-5	19.5#	*Open
CO-V-51		*Open
CO-V-7	27#	*Open
CO-V-8	15#	*Open
DC-V-2A/B	55#	Open
DC-V-65A/B	27#	Closed
DC-V-19A	53#	Open
DC-V-19B	53#	Open
EF-V-30A & D	15#	Closed**
EF-V-30B & C	15#	Closed**
FW-P-1A	27#	*Minimum Gov. Speed
FW-P-1B	27#	*Minimum Gov. Speed
FW-V-16A	27#	*Fail As Is
FW-V-16B	27#	*Fail As Is
FW-V-17A	27#	*Fail As Is
FW-V-17B	27#	*Fail As Is
FW-V-7A	(Later)	*Open
FW-V-7B	(Later)	*Open
IC-V-3	52#	Indeterminate (Note 1)

*Supplied by backup air compressor. Will not go to failed position unless backup air is lost.

**Supplied by two hour back-up bottled air.

Examination Outline Cross-Reference Tier # 1

Evolution/System E02 Vital System Status Verification Group # 1

K/A # EA2.2 Page # 4.3-3 RO/SRO Importance Rating 3.2 3.8

Measurement Ability to operate and / or monitor the following as they apply to the (Vital System Status Verification): Adherence to appropriate procedures and operation within the limitations the facility's license and amendments.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

From the list below, identify the ONE (1) statement that explains why the RCS Loop A/B subcooling margin meters could display values that need to be verified during initial reactor trip transient conditions, so you can ensure transition from EOP-001 to the next (correct) EOP based on the symptom-oriented approach to accident mitigation.

A. The pressure transmitter is slow to respond due to small sensing lines.

B. The temperature transmitter is slow to respond due to the electronics in the RTD preamp.

C. The function generator takes time to make the calculation of Tsat from the pressure signal.

D. The temperature transmitter is slow to respond due to poor thermal coupling between the RCS and the RTD.

Technical Reference Lesson Plan 11.2.01.080, Non-Nuclear Instrumentation, Section T.5, Page 15, Rev. 11.

Open Exam Reference None.

Learning Objective V.E.13.10

Question Source ☐ New ☑ Bank Question # SR5E13-10-Q01

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the pressure transmitter senses changes faster than the temperature transmitter.

Plausible because small sensing lines would result in slower pressure changes resulting in a non-conservative SCM.

B INCORRECT because the slow temperature response is due to poor thermal coupling of the RTD to the RCS.

Plausible because if the electronics did result in slow response, it would give an incorrect SCM resulting in undesired plant transients.

C INCORRECT because the function generator electronics is not the cause for the delay.

Plausible because a delay in the Tsat calculation would result in an inaccurate SCM indication resulting in either a non-conservative reading or undesired plant transients.

D CORRECT. Experience has shown that the RTDs do not respond readily to rapid temperature transients, resulting in in-accurate SCM readings.

Comments Question addresses operation/monitoring the following as they apply to VSSV: Adherence to appropriate procedures and operation within the limitations the facility's license and amendments.

Subcooling margin meters could display erroneous values that could result in incorrect procedure transitions during VSSV. For example, operators are trained to respond to "valid" loss of subcooling margin indications.

Content/Skills	Activities/Notes
2. Digital indication on panel PCL, -100°F to +400°F. Also available from the computer.	
3. Alarmed at < 25°F.	**Obj. 9.4**
4. Calculator generates a saturation temperature from a stored curve using the pressure input. The generated saturation temperature is compared to the temperature input to yield the margin.	
5. Safety Grade Wide Range Th is a different type of RTD than the kind originally installed in the plant (Weeds vs. Baileys), and have a longer time constant. During transients where temperature changes rapidly, such as immediately following a Reactor Trip, these RTDs have been shown to respond more slowly than desired, rendering their input to the Sat. Margin Meter erroneous, and thus the indicated Sat. Margin is also erroneous. A modification to the RTD wells during 7R did not completely eliminate the problem. Current guidance is to not take actions for a Low SCM until it is sure that the instrument is reading properly and not responding to one of its idiosyncrasies.	**Obj. 9.5**
J. Low Range RCS Pressure Detector: 0 - 500 psig, Foxboro Force-Balance Detector. RC3A - PT5	**Obj. 9.1.a, b, c** **PPT 90-91**
1. Detector is mounted on Pressurizer Level instrument tap, LT-3.	
2. Digital indication on PLF.	
3. This is required to be used when RCS pressure is less than 450 psig.	
V. Interim Summary	
1. Point out that so far we have discussed primary side instruments. Review some of the instruments.	**Accelerated Learning Tech.** Have students locate all instrument discussed so far on the 302 and "D" series prints listed in the reference section. Use marked up copy of prints from lesson plan binder to check students.
W. OTSG Pressure Instruments.	**Obj. 9.1.a, b, c** **PPT 92-94**

Examination Outline Cross-Reference Tier # 1

Evolution/System E04 Inadequate Heat Transfer, Loss Secondary Heat Sink Group # 1

K/A # EA1.2 Page # 4.3-8 RO/SRO Importance Rating 3.4 3.8

Measurement Ability to operate and / or monitor the following as they apply to the (Inadequate Heat Transfer) Operating behavior characteristics of the facility.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .5/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Identify the ONE selection below that describes an OTSG that can be declared "available" as a heat sink.

A. RCS is subcooled, with RCPs operating in both loops; Main and Emergency FW are both available; OTSG is dry.

B. Main FW flow is 0 lbm/hr; OTSG tube leak rate is such that no FW flow is required to maintain OTSG level at low level limits.

C. RCS is subcooled with RCPs operating; Main and Emergency FW are both available; OTSG has been isolated because of high level due to OTSG tube leakage.

D. OTSG level and pressure control are possible, but Primary-to-Secondary heat transfer has not been demonstrated; RCS is saturated, with no RCPs operating.

Technical Reference OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 3.10 OTSG Available, Page 4, Rev. 10.

Open Exam Reference None.

Learning Objective IV.E.16.02

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because a dry OTSG is not available as a heat sink IAW OS-24 Section 3.10.

Plausible because under certain circumstances a dry OTSG may be fed IAW EOP-010, Guide 13.

B INCORRECT because OS-24 specifically prohibits level control due to tube leakage.

Plausible because the OTSG has some inventory, potentially allowing it to be a heat sink.

C INCORRECT because OS-24 specifically defines an OTSG isolated per EOP-005 as not available.

Plausible because the OTSG has adequate inventory to serve as a heat sink.

D CORRECT. OS-24 specifically states that primary to secondary heat transfer does not have to be demonstrated to be considered available.

Comments Question addresses ability to operate and/monitor Inadequate Heat Transfer operating behavior characteristics of the facility.

One needs to be able to monitor the correct parameters/conditions during loss of secondary heat sink (EA1.2) in order to be able to mitigate/correct the situation – these all relate to operating behavior characteristics (5), and procedures (10).

3.6 EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER (XHT):

XHT is undesired heat removal by one or both OTSGs. XHT can be confirmed if ALL of the following conditions exist:

- RCS average temperature below 540°F
- Uncontrolled lowering of RCS temperature
- Tsat for OTSG pressure is less than Tcold on affected OTSG(s)

3.7 FEEDWATER:

A water source to the OTSG(s) from either the Main or Emergency Feedwater Systems.

3.8 LACK OF PRIMARY-TO-SECONDARY HEAT TRANSFER (LOHT):

LOHT is the inability of either OTSG to remove sensible heat from the RCS. LOHT can be confirmed if one of the following sets of conditions exists:

- Core exit temperatures rising above 580°F **and** at least one RC Pump operating
- Core exit temperatures rising **and** NO FEEDWATER available
- Core exit temperatures rising **and** RCS circulation can not be confirmed

3.9 MINIMIZE SCM:

An intentional reduction of the reactor coolant pressure temperature relationship as close as practical to the 25°F subcooling margin or RCP NPSH limit. Actions to minimize SCM are described in Guide 8.

3.10 OTSG AVAILABLE:

A physical condition where the OTSG demonstrates level and pressure control. It means the OTSG is in a condition where primary to secondary heat transfer would be possible. Primary to secondary heat transfer need not be demonstrated to determine this availability.

- Primary to secondary leakage should not be considered a means of OTSG level control.
- A dry OTSG is not available.
- An OTSG isolated IAW EOP-005 isolation criteria is not available.

Form ES-401-5 **Written Exam Question Worksheet** Q # 057

Examination Outline Cross-Reference Tier # 1

Evolution/System 003 Dropped Control Rod Group # 2

K/A # AK2.05 Page # 4.2-4 RO/SRO Importance Rating 2.5 2.8

Measurement Knowledge of the interrelations between the Dropped Control Rod and the following: Control rod drive power supplies and logic circuits.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Initial plant conditions:
- Reactor power is 100%, with ICS in full automatic.
- No surveillance testing or maintenance in progress.
- CRD Group 5 is energized from the AUXILIARY power supply.
  - Phases A and B are energized.

Based on these conditions identify the ONE failure below that can cause a SINGLE control rod to drop.

A. ONE (Group 1 Rod 1) CRD motor fuse blows on Phase A.

B. ONE (Group 5 Rod 1) CRD motor fuse blows on Phase A.

C. TWO Auxiliary Power Supply Programmer lamp fuses fail.

D. TWO (Group 5 Rod 1) CRD motor fuses blow on Phases A and B.

Technical Reference CRD Electrical Print 1-TD-006, Rev. 1.

Open Exam Reference None.

Learning Objective V.E.13.06

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # SRO-21 Q-002

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. Loss of phase A will not result in drop of any Safety rod as long as Phase CC is still energized.

Distracter is plausible because during normal operations phases A and CC are energized.

B INCORRECT. Loss of phase one phase will not result in drop of any single rod as long as another phase is still energized.

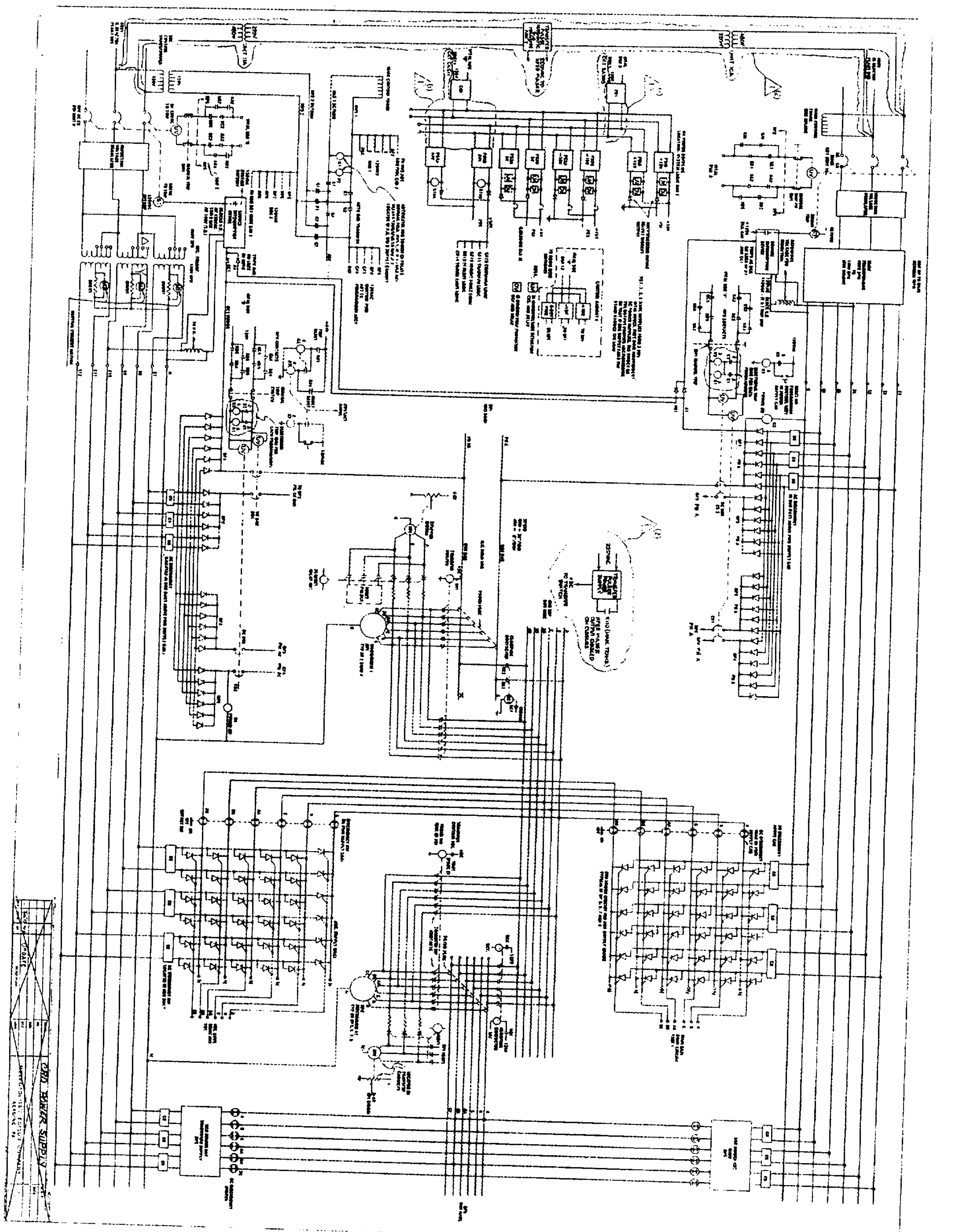
Distracter is plausible because the group is on the auxiliary power supply.

C Incorrect answer. Failure of both programmer lamps wiill result in drop of all rods in that group.

Distracter is plausible because of the credible mis-conception that this failure mode would not result in loss primary and secondary power to the entire group.

D Correct answer. Two fuses blown on the same CRD motor is the only way to drop only a single rod.

Comments Modified May 2003 SRO Exam Q-002.



GPU Nuclear

CRD POWER SUPPLY

MET. ED. CO.
TMI
UNIT 1

DWG. NO. 1-TD-006 REV. 1

SCALE: NONE

TRAINING DRAWING

REV. 0 THIS DWG. IS DEVELOPED FROM MET ED CO. "CRD POWER SUPPLY, SHEET 3, REV 1, DATED 7/10/73".

GPU Nuclear

REV	DRAFT	CHECK	APPROVED	DATE	APP
1	REV. TO INCORPORATE PCN [illegible]				

Form ES-401-5 **Written Exam Question Worksheet** Q # 058

Examination Outline Cross-Reference Tier # 1

Evolution/System 028 Pressurizer Level Malfunction Group # 2

K/A # AA1.06 Page # 4.2-23 RO/SRO Importance Rating 3.3 3.6

Measurement Ability to operate and/or monitor the following as they apply to the Pressurizer Level Control Malfunctions: Checking of RCS leaks.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- RCS T-ave is 579 degrees F, steady.
- Pressurizer level is 218 inches, lowering at 0.5 inches per minute.
- Make up tank level is lowering at 1 inch per minute.
- MU-V-17 is in automatic, with controller output demand at 15%, rising at 2% per minute.
- Indicated RCS makeup flow is 0 gpm.
- Letdown flow is constant at 45 gpm.
- MU-V-32 is in automatic.
- Indicated RCP total seal injection flow is 38 gpm, steady.
- RCP labyrinth seal D/P indicators are normal.
- Auxiliary Building airborne activity is rising.

Based on these conditions, identify the ONE selection below that descibes the cause for this abnormal event.

A. RCS makeup flow is not reaching the RCS.

B. RCS makeup flow transmitter is failing.

C. RCP seal #1 leak-off flow has been isolated by closure of MU-V-26.

D. RCP seal #1 leak-off flow is aligned to the Auxiliary Building sump.

Technical Reference 1203-15, Loss of RC Makeup/Seal Injection, Section 1.0 Symptoms, Page 2, Rev. 28.
302-661, Makeup & Purification System, Rev. 54.

Open Exam Reference None.

Learning Objective IV.A.05.04

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # SR4A05-04-Q01

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer. Symptoms presented are evidence of a leak upstream of the makeup flow sensor.

B INCORRECT. Answer is not consistent with rising Auxiliary Building activity concurrent with the reduction in Makeup Tank level.

Distracter is plausible because indicated makeup flow of 0 gpm with MU-V-17 partially open would support this answer.

C INCORRECT. Answer is not consistent with rising Auxiliary Building activity concurrent with the reduction in Makeup Tank level.

Distracter is plausible because seal leak-off not returning to the Makeup Tank would result in Makeup Tank level lowering.

D INCORRECT. This answer is not supported by the rate of Makeup Tank and Pressurizer level reductions. Total RCP seal #1 leak-off flow is normally 11 gpm. Based on the level reductions in the Makeup Tank (15 gpm) and the Pressurizer (7 gpm) this cannot be the source of the leak.

Distracter is plausible because RCP seal leak-off going to the Auxiliary Building Sump could account for Auxiliary Building airborne activity rising with the corresponding indications of RCS/Makeup Tank leakrate.

Comments Question blends Pressurizer level malfunction with identification of RCS leak location on the RCS Makeup line. It is appropriate to consider Makeup System leakage to be RCS leakage, since Makeup System is governed by RCS leakage Tech Spec.

1.0 **SYMPTOMS**

1.1 Makeup flow indication low as indicated on MU24FI on console "CC".

1.2 MU Pump discharge header pressure high (3100 PSIG) or low (2400 PSIG) as indicated on P.I. MU2 on console "CC".

1.3 Seal injection flow to RCP seals less than 22 gpm as indicated by MU-42 FI on console "CC".

1.4 RCP seal total injection flow Hi/Lo Alarm F-1-5 - Low < 22 gpm.

1.5 Increasing No. 1 Seal Inlet (RC20-TE) and Radial Bearing Temperature. (RC19-TE) computer alarms on PTS A0521-A0528.

> **NOTE**
>
> The conditions in (1.5) may also be an indication of turning vane diffuser bolt failure (Ref. Westinghouse Advisory 88-508).

1.6 RC pump lab seal DP Lo Alarm - F-1-6 (≤ 10") for RCP-1A, 1B, 1D. Less than about 20 inches lab seal ΔP will accelerate deposition of crud on #1 seal (electrophoresis).

2.0 **IMMEDIATE ACTION**

A. Automatic Action

1. Decreasing flow caused MU-V32 RCP seal injection flow control valve to open and decreasing pressurizer level will open MU-V17.

2. Low seal injection flow (< 22 gpm) with low ICCW flow (< 550 gpm) will cause trip of all RCP's after short time delay.

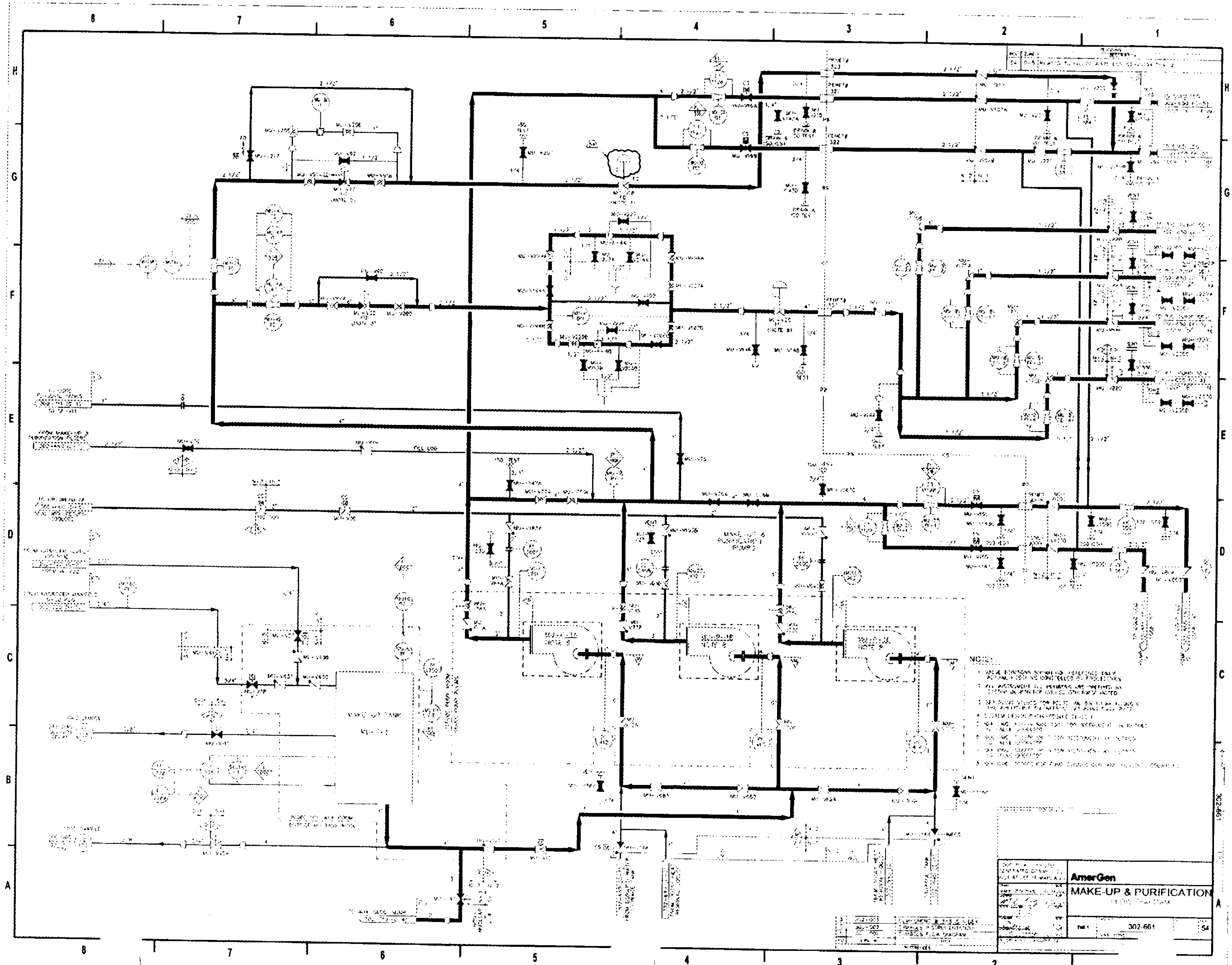
B. Manual Action

1. Determine cause for loss of R.C. Makeup/Seal Injection

   a. Running make up pump trips (Green and amber control switch light)

   b. MU-V17 closed. (Zero flow on MU24FI with MU-P1B operating.)

   c. MU-V32 closed. (Zero flow on MU42FI with MU-P1B operating).



AmerGen
MAKE-UP & PURIFICATION
302-661
54
MAKE-UP TANK
NOTES:

Form ES-401-5 **Written Exam Question Worksheet** Q # 059

Examination Outline Cross-Reference Tier # 1

Evolution/System 060 Accidental Gaseous Radwaste Release Group # 2

K/A # AK3.02 Page # 4.2-46 RO/SRO Importance Rating 3.3 3.5

Measurement Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: Isolation of the auxiliary building ventilation.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .5/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor is operating at 100% power, with ICS in full automatic.
- Waste Gas Compressor separator tank fitting leak is releasing radioactive gas into the compressor room.
- Auxiliary Building Exhaust ventilation fans AHE-14A/C are operating.
- Auxiliary Building Supply Fan AH-E-11 is operating.

Event:
- Radiation Monitor RM-A-6G high alarm actuates.

Based on these conditions, identify the ONE selection below that describes the response of the ventilation system, AND the reason for that response.

A. Auxiliary Building Exhaust fans AH-E-14A/C are tripped to terminate the flow of radioactive gas from the Auxiliary Building.

B. Auxiliary Building Exhaust fans AH-E-14A/C are tripped to terminate flow of radioactive gas from the compressor room to the basement hallway to avoid contamination of the Auxiliary Building.

C. Auxiliary Building supply fan AH-E-11 is tripped to terminate flow of radioactive gas from the compressor room to the basement hallway to avoid contamination of the Auxiliary Building.

D. Auxiliary Building supply fan AH-E-11 is tripped to ensure a negative pressure is established in the Auxiliary Building to avoid release of unmonitored and unfiltered radioactive gas.

Technical Reference Lesson Plan 11.2.01.234, Auxiliary and Fuel Handling Buildings Ventilation System, Pages 8 and 11, Rev. 6.
Radiation Monitoring System Lesson Plan 11.2.01.118, Page 4, Rev. 18.

Open Exam Reference None.

Learning Objective IV.E.06.04

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because RM-A-6G is not interlocked with AH-E-14A/C.

Distracter is plausible because AH-E-14 A/C discharge to the outside atmosphere.

B INCORRECT because RM-A-6G is not interlocked with AH-E-14A/C.

Distracter is plausible because AH-E-14A/C take suction on the Aux building so that the radioactive gas be drawn through the basement hallway.

C INCORRECT because AH-E-11 is shutdown to maintain a negative pressure in the Aux Building. Distracter is plausible because AH-E-11 is interlocked with RM-A-6G.

D CORRECT. AH-E-11 is tripped to maintain a negative pressure in the Aux Building to prevent un-monitored leakage to the outside.

Comments None.

7. Aux/FHB Ventilation System Fire Protection

   See 1104-15A enclosures and alarm panel responses for details on fire protection interlocks.

   The fire protection provided for the Aux/FHB Ventilation System is extensive. It consists of numerous detectors, sensors, alarms, and interlocks.

D. MODES OF OPERATION

1. Startup – general description

   The startup sequence for the Aux/FHB Ventilation System is as follows:

   a. Exhaust (AH-E-14 A/C or B/D)

      1) AH-E-128 must be running or in auto first.

      2) Controls for 14 fans on H and V panel in C.R.

   b. Supply (AH-E-10 and 11)

   c. Recirculation (and cooling) (AH-E-8A or B and AH-E-15 A or B)

      See 1104-15A/B/C for detailed startup requirements
      PPT-26

      1) AH-E-8A/B and AH-E-15A/B may be started and stopped outside the recommended sequence.

      2) AH-E-12 and 13 (331' Aux. Bldg. supply and exhaust fans) also may be started and stopped as desired.

   d. This startup sequence ensures that the Aux/FH Buildings are maintained under a slight negative pressure, thus any leakage is into the buildings versus out of the buildings

      Obj. 4.6

      1) Note: if exhaust fans shutdown, supply fans trip on interlock.

   e. This startup sequence assumes that the radiant heaters, duct heaters, fire protection, air supplies, water supplies, etc., are previously lined up and ready to support the Aux/FHB Ventilation System.

2. Normal Operation

   a. The supply and exhaust fans for the AUX/FH Buildings normally operate continuously. Where standby fans have been provided (except for AH-E-time, the air relief damper will gradually close to a corresponding position.

3) On a further rise in discharge temperature, the electric heating coil will modulate, the outdoor air intake damper will gradually open and the return air damper will gradually close to corresponding positions. On falling temperatures, the reverse of this operation occurs.

3. Shutdown

Obj. 4.6

While it is intended that the Aux/FHB Ventilation System will run continuously, even when the plant is shutdown, there may be times when the system must be shutdown.

a. The normal shutdown sequence is:

See OP 1104-15A for shutdown specifics

1) Supply

2) Exhaust

3) Recirculation Units - may be S/D at any time

b. This sequence will assure that a negative pressure will be maintained within the buildings as long as possible. Operation of the supply components without simultaneous operation of the exhaust components will result in a positive building pressure which could force contaminants out of building openings directly to the atmosphere without having been passed through the filters

INTERIM SUMMARY - The Aux/FHB ventilation system is composed of fans, dampers, heaters, coolers, and ducting all designed to maintain building ambient temperatures within component design limits. The design and operation of the fans is such that a negative building pressure is maintained in order to prevent unmonitored radioactive gas releases to environment.

E. Instrumentation, Controls, Alarms and Protection Devices

1. Group A (Supply)

microcuries/cc was provided in the Auxiliary and Fuel Handling Building exhaust and in the Reactor Building purge exhaust and in Condenser Offgas exhaust.

j. The sampling stations (MAP 5) have been specified to be equipped with 3 filter holders. Each filter is equipped with a solenoid valve controlled by a solid state timer.

k. The bases for this arrangement is to provide one filter to be used for continuous sampling while the other 2 filters are used to obtain a composite sample based on 1/10 and 1/100 sampling time interval.

l. This feature increases the potential for having a filter capable of being analyzed without undo exposure to personnel or with a radioactivity level beyond the capability of standard count room MEA equipment.

5. UFSAR descriptions and responses to NRC questions, while not design basis, are design commitments and have influenced some aspects of the design of the RMS.

   a. Interlocks were provided to enhance negative pressure in buildings upon detection of a high radiation condition in either the Auxiliary or Fuel Handling Building exhaust.

   b. The radiation monitoring channels were assigned 120 volt power from the four vital busses to provide further assurance that one inverter failure does not result in complete RMS failure.

   c. Relationship of sensitivities and ranges to design basis accidents of the RMS were reviewed to assure that they adequately supported the accident analyses.

   d. TMI-1 installed an independent engineered safety features filter system at the TMI-1 fuel handling floor area for postulated fuel handling accidents, with the Fuel Handling Building automatically isolated from the Auxiliary Building by closure or leak tight dampers on either detection of differential pressure

Form ES-401-5 **Written Exam Question Worksheet** Q # 060

Examination Outline Cross-Reference Tier # 1

Evolution/System 076 High Reactor Coolant Activity Group # 2

K/A # AA1.04 Page # 4.2-59 RO/SRO Importance Rating 3.2 3.4

Measurement Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Plant startup in progress.
  - Reactor power is 22%, with ICS in automatic.

Event:
- Operator adjusts ICS control to effect a power increase to 40%.
- Significant rise in Main Condenser off-gas radiation monitor readings with no corresponding symptoms of increasing OTSG tube leak rate.

Based on these conditions, identify the ONE selection below that describes the cause for this event, and the expected automatic response.

A. Increase in RCS activity.
After a time delay due to letdown and sample transport time, RM-L-1 Lo will close letdown isolation valves MU-V-2A and MU-V-2B.

B. Increase in RCS activity.
After a time delay due to letdown and sample transport time, RM-L-1Hi will close letdown isolation valves MU-V-2A and MU-V-2B.

C. Radioactive crud burst on the secondary side of an OTSG.
Actuation of RM-A-5 or RM-A-15 high alarm will start the condenser off-gas MAP-5 iodine sampler.

D. Radioactive crud burst on the secondary side of an OTSG.
Actuation of RM-G-25 (RM-A-5G HI-HI) high alarm will start the condenser off-gas MAP-5 iodine sampler.

Technical Reference Lesson Plan 11.2.01.118, Radiation Monitoring System, PPT # 54, Rev 18.
MAP C-1-1, RM-L-1, Page 42, Rev. 33.

Open Exam Reference None.

Learning Objective IV.E.06.04

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because RM-L-1(low) is not interlocked with MU-V-2A/B.

Distracter is plausible because the other source for steam line and condenser off-gas radiation is failed fuel.

B CORRECT. RCS activity is the alternate source for steam line and condenser off-gas radiation and RM-L-1(HI) is interlocked with MU-V-2A/B.

C INCORRECT because the secondary steam side is a non-radioactive system with only a small possibility of contamination that would not result in the significant rise in readings given in the stem.

Distracter is plausible because an alarm condition on RM-A-5/15 will start the condenser off-gas MAP-5 Iodine Sampler.

D INCORRECT because RM-G-25 is not interlocked with the Condenser Off-Gas MAP-5 Iodine Sampler.

Distracter is plausible because if a radioactive crud burst of sufficient magnitude did occur on the secondary side of the OTSG, the Steam Line and Condenser Off-Gas monitors would rise.

Comments None.

Liquid Monitors
Interlocks

- RM-L-1 HI -
  - Close MU-V-2A / 2B (isolate letdown)
- RM- L-6 -
  - Close WDL-V-257 (stop liquid release)
- RM- L-7 -
  - Close WDL-V-257 (stop liquid release)
- RM- L-12 -
  - Trip IW-P-16,17,18,29,30 and Close IW-V-73 and IW-V-279 (stop IWTS/FS release)

C-1-1
Revision 33

ALARM:

RM-L-1 (PRIMARY COOLANT LETDOWN)

SET POINTS:

Refer to OP 1101-2.1, RMS Setpoints.

CAUSES:

High primary coolant activity

AUTOMATIC ACTION:

MU-V-2A and 2B close on Hi Alarm on RM-L-1 Hi.

OBSERVATION (CONTROL ROOM):

1. RM-L-1 "Alert" Alarm on PRF
2. RM-L-1 "Hi Alarm" on PRF
3. RM-L-1 Indication on PRF > above setpoints

MANUAL ACTION REQUIRED:

NOTE
There is a delay time between changes in RCS activity and RML 1 response of 30 to 60 minutes depending on RM-L-1 flow rate.

1. Verify MU-V2A and 2B closed if Hi Alarm on RM-L-1 Hi.
2. Verify Hi RCS Activity (i.e.: RM-L-1 Lo, RCS Sample). If valid Hi RCS Activity, then do not re-establish letdown thru MU-V2A and 2B. This will help insure radiation levels in the Aux. Building and component reliability with respect to exposure stay within design limits.
3. Refer to EP 1202-11, High Activity in Reactor Coolant.
4. Refer to EP 1202-12, Excessive Radiation Levels.

Form ES-401-5 **Written Exam Question Worksheet** Q # 061

Examination Outline Cross-Reference Tier # 1

Evolution/System A02 Loss of NNI-X Group # 2

K/A # AK3.3 Page # 4.3-29 RO/SRO Importance Rating 3.7 3.2

Measurement Knowledge of the reasons for the following responses as they apply to the (Loss of NNI-X): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

10CFR55.41(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .5/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** C.

Plant conditions:
- Total loss of ICS Hand and Auto Power.
- Reactor tripped.
- NO required immediate manual actions have been completed yet.

Based on these conditions identify the ONE selection below that describes the impact of re-energizing ICS/NNI power prior to completion of the pre-restoration actions.

A. Loss of RCP seal injection due to closure of MU-V-32.

B. RCS pressure and Pressurizer level excursion due to MU-V-17 opening.

C. OTSG overfeed condition due to repositioning of the Main Feedwater valves.

D. Excessive RCS cooldown rate due to repositioning of atmospheric dump valves MS-V-4A/B.

Technical Reference 1202-40, Caution at Step 2.2, Page 3, Rev. 41.

Open Exam Reference None.

Learning Objective IV.E.27.44

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because loss of ICS/NNI Auto and Hand Power causes MU-V-32 to fail to mid-scale, thus maintaining seal injection to the RCPs.

Distracter is plausible because some valves travel closed on restoration of ICS/NNI power.

B INCORRECT because MU-V-17 fails to the mid-position and will remain there on restoration of power.

Distracter is plausible because some valves travel to their full open position on restoration of power, e.g., FW-V-16/17 valves.

C CORRECT. FW-V-17A/B and FW-V-16A/B travel to their full open positions on restoration of power.

D INCORRECT because MS-V-4A/B shift to the backup loader and remain closed.

Distracter is plausible because some valves travel to their full open position on restoration of power, e.g., FW-V-16/17 valves.

Comments This question addresses reason for not manipulating specific controls in order to avoid undesirable operating results. The procedure directs the operators NOT to do this so we get do not get the overfeed condition. Overfeed event would produce excessive RCS cooldown.

2.2 Manual Action

> **CAUTION**
>
> Do not select alternate ICS/NNI Power or otherwise attempt to restore power at this point. Upon restoration of HAND power, main and startup feedwater valves will stroke fully open.

> **CAUTION**
>
> If a. or b. below cannot be performed as written (i.e., ATWS or failure of main turbine stop valves to close) go directly to OP-TM-EOP-001 at that point for direction on performance of remedial actions. Refer to this procedure for additional guidance.

On a confirmed loss of ICS Hand and ICS Auto power.

1. **TRIP** the reactor and **VERIFY** power less than 10%.
2. **TRIP** the main turbine and **VERIFY** T/G stop valves closed.
3. **TRIP** both main feedwater pumps.
4. **GO TO** OP-TM-EOP-001 and refer to this procedure for additional guidance.

> **NOTE**
>
> Control room indications listed in Table 1, are available for controlling plant parameters.

3.0 **<u>FOLLOW UP ACTION</u>**

Objective: The objective of this procedure is to stabilize the plant in a hot shutdown condition and to restore ICS/NNI power. If unable to restore power, proceed with a controlled plant cooldown.

_____ A. **VERIFY** EFW Controls OTSG Level at ≥ 25" startup range.

_____ B. **OPEN** MS-V-4A/B with B/U loaders, ("BACK UP CTRL" Bailey Stations) to reseat main steam safety valves and control OTSG pressure.

C. **IF** MU-V-17 cannot be controlled in Hand or Auto, **THEN** (NA if MU-V-17 can be controlled)

_____ a. **USE** MU-V-217 to control pressurizer level.

_____ b. **DISPATCH** an operator to isolate MU-V-17 locally **BY CLOSING MU-V-91B**.

Examination Outline Cross-Reference Tier # 1

Evolution/System A06 Shutdown Outside Control Room Group # 2

K/A # 2.1.23 Page # 2-3 RO/SRO Importance Rating 3.9 4.0

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

10 CFR Part 55 Content ☐ 55.41 .... ☐ 55.43 ....

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor is operating at 100% power with ICS in full automatic.
- Fire conditions require Control Room to be evacuated.
  - NO immediate actions of OP-TM-EOP-020, Cooldown From Outside of Control Room, were performed prior to the evacuation.

Based on these conditions, identify the ONE selection below that describes how OP-TM-EOP-020 directs you to TRIP THE REACTOR.

A. Open CRD Breakers 10 and 11.

B. Open CRD breakers at 1G and 1L 480V switchgear.

C. Pull the Main Turbine Trip handle at the Front Standard.

D. Pull the Trip Handles at FW-P-1A and FW-P-1B local control consoles.

Technical Reference OP-TM-EOP-020, Cooldown From Outside Of Control Room, Pages 1 and 63, Rev. 3.

Open Exam Reference None.

Learning Objective V.D.18.02

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because OP-TM-EOP-020 attachment 12 directs the CRD breakers at the 1G and 1L busses to be tripped if the reactor was not tripped prior to leaving the Control Room.

Distracter is plausible because opening CRD 10 and 11 breakers will cause a reactor trip.

B CORRECT because OP-TM-EOP-020 attachment 12 directs the CRD breakers at the 1G and 1L busses to be tripped if the reactor was not tripped prior to leaving the Control Room.

C INCORRECT because OP-TM-EOP-020 attachment 12 directs the CRD breakers at the 1G and 1L busses to be tripped if the reactor was not tripped prior to leaving the Control Room.

Distracter is plausible because OP-TM-EOP-020 attachment 12 directs pulling the main turbine trip handle to trip the main turbine. It is also plausible because tripping the main turbine will also cause the reactor to trip.

D INCORRECT because OP-TM-EOP-020 attachment 12 directs the CRD breakers at the 1G and 1L busses to be tripped if the reactor was not tripped prior to leaving the Control Room.

Distracter is plausible because OP-TM-EOP-020 attachment 12 directs pulling the main feedwater pump trip handles to trip the main feed pumps. It is also plausible because tripping the main feed pumps will also cause the reactor to trip.

Comments None.

COOLDOWN FROM OUTSIDE OF CONTROL ROOM

1. **ENTRY CONDITIONS** - Fire in the relay room or Control Room, or another hazard which threatens to make the Control Room uninhabitable or threatens the ability to achieve safe shutdown from the Control Room.

2. **IMMEDIATE ACTIONS**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 2.1 **TRIP** the reactor	
___ 2.2 **PERFORM** OP-TM-EOP-001 Immediate Manual Actions.	**GO TO** Attachment 12.
___ 2.3 **INITIATE** OP-TM-EOP-010 Guide 15 "EFW Actuation Response".	**GO TO** Attachment 12.
___ 2.4 **TRIP** both main Feedwater Pumps.	**GO TO** Attachment 12.
___ 2.5 **TRIP** all reactor coolant pumps.	**GO TO** Attachment 12.
___ 2.6 **OPEN** MU-V-14A **and** MU-V-14B.	
___ 2.7 **CLOSE** RC-V-2.	

3. **FOLLOW-UP ACTIONS**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.1 Stabilize at Hot Shutdown	
___ 3.1.1 **ANNOUNCE** "reactor trip" and "______ requires commencing remote shutdown sequence", over the plant page and radio.	___ **PERFORM** Step 3.1.1 after Step 3.1.2 (communications isolated from control & relay rooms)

NOTE:

(1) The typical assignments for a cooldown outside of the CR are as follows. SM, STA & URO go to the control tower 2nd floor and then to the RSD panels, US & ARO go to the control tower 3rd floor and then to the RSD panels, Secondary reading AO goes to the EFW area, and Primary AO goes to MU valve alley.

(2) M&I phones should be utilized for communications between all of these locations. Cross tie primary and secondary M&I channels.

(3) Portable emergency lighting is available at RSD area, Maintenance Shop East of Roll Up Door, RB hatch and Primary AO Central

ATTACHMENT 12
Contingencies for Control Room Actions
Page 1 of 1

NOTE:	The following actions assume that Control Room evacuation is required immediately. If time is available, alternate means to accomplish these actions from the Control Room should be considered.

1. **If** reactor was not tripped,
   then OPEN CRD breakers 1G-2A and 1L-2A (CB 322: patio). ____
2. **If** turbine was not tripped,
   then PULL "TRIP" handle for Main Turbine (TB 355: North end of turbine). ____
3. **If** EFW was not initiated, **then PLACE** EF-P-2A **and** EF-P-2B 69 switch in EMERG **and CLOSE** breakers at 4160V bus 1D & 1E. ____
4. **If** both Main FW Pumps were not tripped, **then**

   PULL "TRIP" handle for FW-U-1A (TB 322: FW-U-1A control console). ____

   PULL "TRIP" handle for FW-U-1B (TB 322: FW-U-1B control console). ____
5. **If** all RC pumps were not tripped, then at 6900V switchgear (TB 322)

   UNLOCK and PLACE RC-P-1A 69 switch in EMERG (69 key required). ____

   TRIP 1A 6900V Unit 1A2 (RC-P-1A breaker). ____

   UNLOCK and PLACE RC-P-1C 69 switch in EMERG (69 key required). ____

   TRIP 1A 6900V Unit 1A3 (RC-P-1C breaker). ____

   UNLOCK and PLACE RC-P-1B 69 switch in EMERG (69 key required). ____

   TRIP 1B 6900V Unit 1B2 (RC-P-1B breaker). ____

   UNLOCK and PLACE RC-P-1D 69 switch in EMERG (69 key required). ____

   TRIP 1B 6900V Unit 1B3 (RC-P-1D breaker). ____
6. **INITIATE** OP-TM-EOP-010 Guide 15 "EFW Actuation Response". ____
   - To start EF-P-1: **CLOSE** IA-V-1129 (IB 295: IA supply to MS-V-13A) **and OPEN** regulator bleed valve to OPEN MS-V-13A **or CLOSE** IA-V-1133 (IB 295: IA supply to MS-V-13B) **and OPEN** regulator bleed valve to OPEN MS-V-13B.
7. **GO TO** the follow-up actions. ____

Form ES-401-5 **Written Exam Question Worksheet** Q # 063

Examination Outline Cross-Reference Tier # 1

Evolution/System 036 Fuel Handling Incidents Group # 2

K/A # AK2.01 Page # 4.2-28 RO/SRO Importance Rating 2.9 3.5

Measurement Knowledge of the interrelations between the Fuel Handling Incidents and the following: Fuel handling equipment.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** A.

Identify the ONE selection below that describes how TMI prevents the ability to INSERT or RAISE an irradiated fuel assembly in the New Fuel Elevator.

A. Administrative controls.

B. New Fuel Elevator weight sensing interlock.

C. Fuel Handling overhead crane cable length restriction.

D. Bridge Dillon cell interlock associated with elevator approach corridor.

Technical Reference Lesson Plan 11.2.01.265, Fuel Handling Equipment, Page 4, Rev. 13.
1507-6, New Fuel Elevator Operation, Limit & Precaution 5.5, Page 3, Rev. 7.

Open Exam Reference None.

Learning Objective IV.B.15.05

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. There is no equipment interlock to prevent this specific incident.

B INCORRECT. There is no equipment interlock to prevent this specific incident.

Distracter is plausible because an overload interlock would prevent raising any assembly, irradiated or not, in the new fuel elevator.

C INCORRECT. There is no equipment interlock to prevent this specific incident.

Distracter is plausible because other plants have physical limits which would prevent grappling an assembly inserted down into the fuel storage racks.

D INCORRECT. There is no equipment interlock to prevent this specific incident.

Distracter is plausible because Dillon cells are used on the bridge masts to provide automatic (up direction) stop, set at specific weights to prevnet damage if an assembly "hangs up" as it is being raised.

Comments Question addresses how we PREVENT fuel handling incidents while operating spent fuel handling equipment. The fact that there is no equipment interlock to prevent this specific incident raises the required level of safety concern for all equipment operators, and the importance of this issue as part of the operator licensing process.

Content/Skills	Activities/Notes
b. Mechanical stops provided in case of geared **limit switch** failure. These stops hold the elevator on the rails.	
1) Upper stop stalls motor 2) Lower stop stops elevator in down position.	**In both cases the NFE stays on its rails.**
c. Cable is marked with a reference mark to ensure elevator is fully down.	
6. Brakes	**Obj. 15.04**
a. Hoist motor has a disc type brake associated with it. This brake operates when a solenoid de-energizes when the hoist motor is de-energized by the limit switches or the stop PB.	
b. A mechanical load brake is also provided to control lowering speed and to prevents the load from dropping rapidly due to electric brake failure. This is a ratchet, pawl and friction type load brake.	
7. **<u>Limit and Precaution about Irradiated Fuel and New Fuel Elevator:</u>**	**Obj. 15.05**
a. RP 1507-6 Step 5.5 "Irradiated Fuel assemblies **SHALL NOT** be placed in the New Fuel Elevator:"	
1) ALARA 2) Personal Safety	No interlock prevents putting an irradiated fuel assembly in the elevator and raising it. Question: What does **ALARA** stand for? **A**s **L**ow **A**s **R**easonably **A**chievable
Interim Summary:	
Review: a. Why we have New Fuel Elevator b. Location of New Fuel Elevator c. Safety precautions around Spent Fuel Pool. d. Go over any applicable OE	**Review Objectives using questioning techniques to measure effectiveness of training**

1.0 **PURPOSE**

The purpose of this procedure is to provide instructions for the operation of the new fuel elevator.

2.0 **DESCRIPTION**

The new fuel elevator is used to vertically transport new fuel assemblies from the 348' operating floor of the Fuel Handling Building to the fuel storage rack level. The new fuel elevator is designed such that a new fuel assembly can be grappled by the spent fuel handling bridge for transportation to either fuel storage racks or the fuel transfer system upenders.

NOTE
Enclosure 1 provides additional equipment information. Typically, the new fuel elevator is used in conjunction with 1503-1 (Ref. 3.3) activities.

3.0 **REFERENCES**

3.1 1303-11.4, Refueling System Interlocks

3.2 Instruction Manual - New Fuel Elevator Stearns-Roger Corporation

3.3 1503-1, Receipt of New Fuel and Control Components

3.4 1507-2, Fuel Handling Building Crane Operation

4.0 **TOOLS, EQUIPMENT AND SUPPLIES**

None

5.0 **LIMITS AND PRECAUTIONS**

5.1 The elevator must be visually observed at all times when it is in motion.

- The operator shall be prepared to stop the hoist immediately if any malfunction should be evidenced. Close attention is required when approaching the limit switches.

5.2 When a fuel assembly, rod handling container or dummy fuel assembly is in the elevator attached to a sling, no horizontal movement of the service crane is permitted (Ref. 3.4).

5.3 The elevator must always be stored in the "DOWN" position when not in use.

5.4 The operator must verify the "DOWN" position by checking that the reference mark on the cable lines up with the mark on the winch (±2 inches).

5.5 Irradiated fuel assemblies **SHALL NOT** be placed in the new fuel elevator.

5.6 Verify that the elevator is in the "DOWN" position prior to moving the spent fuel handling bridge over the elevator.

Form ES-401-5 **Written Exam Question Worksheet**

Examination Outline Cross-Reference

Tier # 1

Group # 2

Evolution/System E03 Inadequate Subcooling Margin

K/A # EA2.2 Page # 4.3-6 RO/SRO Importance Rating 3.5 4.0

Measurement Ability to determine and interpret the following as they apply to the (Inadequate Subcooling Margin): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 .... ☑ 55.43 .5

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor tripped.
- RCS LOCA in progress.
- Train A and Train B HPI Actuation.

Event:
- Loss of RCS subcooled margin.
- URO stops RC-P-1A, RC-P-1B, and RC-P-1C.
- RC-P-1D would NOT trip from the main control panel within the procedurally required time limit.

Based on these conditions identify the ONE selection below that describes actions required at this time.

A. Restart all available RCPs.

B. De-energize 1B 6900V Bus.

C. Restart RC-P-1A or RC-P-1B.

D. Continue to operate RC-P-1D.

Technical Reference OP-TM-EOP-010 Rule 1, SCM, Page 4, Rev. 10.

Open Exam Reference None.

Learning Objective V.E.10.3

Question Source ☐ New ☐ Bank ☑ Modified Bank

Question #

Parent Question # QR-PCO-04-EOP002-Q01

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because EOP-010, Rule 1 response not obtained column requires leaving only un-tripped RCP(s) running.

The purpose of tripping the RCPs is to reduce two phase flow out of the break before system void fraction becomes high enough that if phase separation occurred (last pump was tripped too late) the core would be uncovered. It is plausible that starting all RCPs at this time would ensure phase separation did not occur, followed by core uncovery.

B INCORRECT because EOP-010, Rule 1 response not obtained column requires leaving only un-tripped RCP(s) running.

Distracter is plausible because the required action if all RCPs do not turn off is to de-energize the associated 7 KV bus if within 1 minute of loss of subcooling margin.

C INCORRECT because EOP-010, Rule 1 "Response Not Obtained" column requires leaving only un-tripped RCP(s) running.

Distracter is plausible because starting RC-P-1A or RC-P-1B would give balanced flow with 1 RCP operating in each loop.

D CORRECT action IAW EOP-010, Rule 1, response not obtained.

Comments Modified TMI Bank - QR-PCO-04-EOP002-Q01.

SCM

1

Rule 1
Loss of Subcooling Margin (SCM)

IAAT SCM < 25°F **and** reactor is shutdown, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** it has been more than two minutes since RCP start	**GO TO** Step 3
2. **ENSURE** all RCPs are shutdown.	**If** all RCPs were **not** tripped within one minute, **then MAINTAIN** RCP(s) still operating until one of the following conditions is satisfied: SCM > 25F LPI flow > 1250 gpm/leg Tclad > 1800°F
3. **ENSURE** 1600 # ESAS has been actuated	**PERFORM** Guide 2 "HPI/LPI Initiation"
4. **ENSURE** EFW has actuated.	
5. **VERIFY** all HPI and LPI components are in the ES condition.	**INITIATE** Guide 3 (LPI) or Guide 4 (HPI).
6. **INITIATE** Guide 15 **and FEED** available OTSGs to 75 to 85% Operating Range Level.	

Examination Outline Cross-Reference Tier # 1

Evolution/System E13 EOP Rules Group # 2

K/A # EK1.2 Page # 4.3-21 RO/SRO Importance Rating 3.0 3.6

Measurement Knowledge of the operational implications of the following concepts as they apply to the (EOP Rules) Normal, abnormal and emergency operating procedures associated with (EOP Rules).

10CFR55.41(8) Components, capacity, and functions of emergency systems.
10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .8/.10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** D.

Plant conditions:
- PORV HPI cooling is in progress.
- RCS subcooled margin is 10 degrees F.
- RCPs are all tripped.
- Both OTSG pressure boundaries are intact.

	Level	Pressure	Tube-Shell DT
- OTSG 1A	9 inches	240 psig	+62 degrees
- OTSG 1B	7 inches	210 psig	+64 degrees

- Core exit thermocouple temperatures are now steady, no longer rising.
- There are no symptoms of OTSG tube leakage.

Emergency Feedwater (EFW) is now available to re-establish FW to the OTSGs.

Based on these conditions, identify the ONE selection below that describes OP-TM-EOP-010 required actions.

A. Establish EFW to OTSG 1A ONLY, at greater than 185 gpm but less than 430 gpm.

B. Establish EFW to OTSG 1A ONLY, at greater than 430 gpm.

C. Establish EFW to BOTH OTSGs at a MAXIMUM of 435 gpm to each OTSG.

D. Establish EFW to BOTH OTSGs, at greater than 215 gpm to each OTSG with NO upper limit.

Technical Reference OS-24, Section 3.10, Page 4, Rev. 10.
OP-TM-EOP-010 Rule 4, Feedwater Control, Page 8, Rev. 3.
OP-TM-EOP-010 Guide 13, Feeding a Dry OTSG, Page 28, Rev. 3.
OP-TM-EOP-010 Guide 14, Tube-to-Shell Delta T Limit/Control, Page 29, Rev. 3.

Open Exam Reference None.

Learning Objective V.E.10.3

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because both OTSGs are available IAW OS-24, therefore both OTSGs will be fed.

Distracter is plausible because the feed rate given in the second part is correct for a DRY OTSG without RCPs running. The status of RCPs was given in the stem. Also, OTSG 1A has the lower tube to shell delta T, making it the safer OTSG to feed.

B INCORRECT because both OTSGs are available IAW OS-24, therefore both OTSGs can be fed.

Distracter is plausible because the feed rate given is correct for only 1 OTSG available with sub cooling margin less than 25°F.

C INCORRECT because the feed rate given is for a dry OTSG and neither OTSG is less than 6" as given in the stem.

Distracter is plausible because both OTSGs are available IAW with OS-24, therefore both OTSGs can be fed.

D CORRECT because:
(1) Both OTSGs are AVAILABLE in accordance with OS-24 section 3.10.
(2) Rule 4: With both OTSGs available and SCM <25 degrees, establish >215 gpm/OTSG.

Comments None.

3.6 EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER (XHT):

XHT is undesired heat removal by one or both OTSGs. XHT can be confirmed if ALL of the following conditions exist:

- RCS average temperature below 540°F
- Uncontrolled lowering of RCS temperature
- Tsat for OTSG pressure is less than Tcold on affected OTSG(s)

3.7 FEEDWATER:

A water source to the OTSG(s) from either the Main or Emergency Feedwater Systems.

3.8 LACK OF PRIMARY-TO-SECONDARY HEAT TRANSFER (LOHT):

LOHT is the inability of either OTSG to remove sensible heat from the RCS. LOHT can be confirmed if one of the following sets of conditions exists:

- Core exit temperatures rising above 580°F **and** at least one RC Pump operating
- Core exit temperatures rising **and** NO FEEDWATER available
- Core exit temperatures rising **and** RCS circulation can not be confirmed

3.9 MINIMIZE SCM:

An intentional reduction of the reactor coolant pressure temperature relationship as close as practical to the 25°F subcooling margin or RCP NPSH limit. Actions to minimize SCM are described in Guide 8.

3.10 OTSG AVAILABLE:

A physical condition where the OTSG demonstrates level and pressure control. It means the OTSG is in a condition where primary to secondary heat transfer would be possible. Primary to secondary heat transfer need not be demonstrated to determine this availability.

- Primary to secondary leakage should not be considered a means of OTSG level control.
- A dry OTSG is not available.
- An OTSG isolated IAW EOP-005 isolation criteria is not available.

FWC

4

Rule 4
Feedwater Control

A. **IAAT** the reactor is shutdown, **then:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** SCM > 25°F.	**MAINTAIN** OTSG level 75 – 85% OPERATING Range Level.
2. **VERIFY** at least 1 RCP operating.	**MAINTAIN** OTSG level ≥ 50% OPERATING Range Level.
3. **MAINTAIN** OTSG level ≥ 25" STARTUP Range Level.	

B. **IAAT** OTSG Level < minimum, **then MAINTAIN** the following MINIMUM required flow:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **If** SCM < 25°F **and** <u>both</u> OTSGs are available, **then FEED** > 215 gpm/OTSG using EFW.	**FEED** > 1.0 Mlbm/hr using MFW.
2. **If** SCM < 25°F **and** <u>only one</u> OTSG is available, **then FEED** > 430 GPM to the good OTSG using EFW.	**FEED** > 1.0 Mlbm/hr using MFW.
3. **If** all RCPs are OFF **and** incore temperature is rising**, then FEED** OTSG at maximum available EFW flow.	**FEED** > 1.0 Mlbm/hr using MFW.
4. There is no minimum required flow rate.	

Guide 13
Feeding a Dry OTSG

IAAT OTSG SU Level < 6" **and** OTSG pressure at least 200 psi below P_{sat} for $T_{c,,}$ and feedwater is available, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** feeding the DRY OTSG is required for adequate core cooling **or** TSDT is approaching limits.	**OBTAIN** direction from TSC to feed the DRY OTSG.

NOTE
(1) Automatic EFW actuation is not restricted by this guidance.
(2) RCP operation is desired.
(3) Feeding a DRY OTSG takes precedence over Tube to Shell ΔT required actions (Guide 14) while OTSG pressure is being restored.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. **VERIFY** the DRY OTSG pressure boundary is INTACT.	1. **VERIFY** all RCPs are OFF **or** TSDT Limits are being challenged 2. **VERIFY** the OTSG pressure boundary failure is **not** in the Intermediate or Reactor Building.
3. **If** TSDT is negative, **then,** 1) **If** OTSG pressure boundary is **not** intact, **then** **VERIFY** an RCP is operating. 2) **FEED** the DRY OTSG at a maximum flow of 0.1 Mlbm/HR using Main Feedwater.	**If** RCPs are OFF, **then FEED** the DRY OTSG at a maximum of 185 GPM using EFW.
4. **If** TSDT is positive, **then** **FEED** using EFW 1) **If** at least one RCP is ON, **then** the maximum flow is 435 GPM 2) **If** RCPs are OFF, **then** the maximum flow is 185 GPM	
5. **When** OTSG pressure is within 200 psig of Psat for Tcold, **then** these feedwater flow limits no longer apply.	

Guide 14
Tube-to-Shell Delta-T Limit/Control

NOTE

(1) Negative TSDT (tube to shell differential temperature) means the tubes are colder than the shell.

(2) Positive TSDT means the tubes are hotter than the shell.

IAAT the reactor is shutdown, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** OTSG tube-to-shell differential temperature (TSDT) (as indicated on PPC points C4015 and C4016) is above (less negative) the tensile limit, **–70°F**.	**PERFORM** actions in order listed until TSDT is controlled above limit: 1. **MINIMIZE** SCM. 2. **REDUCE** cooldown rate **or** as necessary, hold or raise RCS temperature. 3. **FEED** the OTSG with MFW versus EFW to enhance shell cooling. • **USE** Guide 13 if OTSG is dry.
2. **VERIFY** OTSG tube-to-shell differential temperature (TSDT) (as indicated on PPC points C4015 and C4016) is below the compressive limit **+60°F**.	**PERFORM** actions in order listed until TSDT is controlled below limit: 1. **REDUCE** heatup rate **or** as necessary, hold or lower RCS temperature. 2. **FEED** the OTSG with EFW versus MFW to enhance tube cooling. • **USE** Guide 13 if OTSG is dry. 3. **TRIP** the reactor coolant pumps in the affected loop
3. **If** RCS Temperature > 500°F **and** RCS Pressure < 1800 psig**, then VERIFY** OTSG tube-to-shell differential temperature (TSDT) (as indicated on PPC points C4015 and C4016) is below the compressive limit **+50°F.**	**PERFORM** actions in order listed until TSDT is controlled below limit: 1. **REDUCE** heatup rate **or** as necessary, hold or lower RCS temperature. 2. **FEED** the OTSG with EFW versus MFW to enhance tube cooling. • **USE** Guide 13 if OTSG is dry. 3. **TRIP** the reactor coolant pumps in the affected loop.
4. **If** any of the limits above are approached, **then NOTIFY** the TSC to review the TSC Guidelines for additional tube-to-shell delta-T control options.	

Form ES-401-5 **Written Exam Question Worksheet** Q # 066

Examination Outline Cross-Reference Tier # 3

Evolution/System Conduct of Operations Group #

K/A # 2.1.29 Page # 2-4 RO/SRO Importance Rating 3.4 3.3

Measurement Knowledge of how to conduct and verify valve lineups.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Maintenance work completed on DC-V-2A, DHR Heat Exchanger Inlet.
- Control Room Supervisor requests independent verification that DC-V-2A is in its ES required position.
- A qualified operator is ready to verify locally that DC-V-2A is open.

Identify the ONE selection below that describes required actions for a second individual to complete this type of verification.

A. Verify the Control Room ES Status Panel keyswitch is in the normal operating position.

B. Verify DC-V-2A is open from its controller in the Control Room.

C. Visually observe the first individual checking DC-V-2A open.

D. Independently verify locally that DC-V-2A is physically open.

Technical Reference HU-AA-101, Human Performance Tools and Verification Practices, Sections 4.3.2.4 and 4.3.2.5, Pages 7and 8, Rev. 2.

Open Exam Reference None.

Learning Objective V.J.01.02

Question Source ☐ New ☑ Bank Question # Audit 20 Q087

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because position of this keyswitch does not ensure the valve is actually open.

Distracter is plausible because this switch de-energizes solenoids to block, bleed air from DC-V-2A and DC-V-65A.

B INCORRECT because this is not an acceptable method of position verification due to being a demand signal, not a true position indication.

Distracter is plausible because the controller is in the control room and would show the demanded position of DC-V-2A.

C INCORRECT because this is not a true independent check of DC-V-2A valve position.

Distracter is plausible because this method is similar to the method used where verification of throttle valve setting is required.

D CORRECT. Proper procedure for true independent valve position verification IAW HU-AA-101.

Comments 2001 Audit Q-087.

4.3.2.2. Independent Verification (IV) may be applicable to -but not limited to- component manipulations; Clearance application / removal, and performance of a procedure or other activities that will remove/restore equipment from/to service. Examples are:

1. Re-Landing Leads
2. Relay contact boot removal
3. Jumper removal
4. Valve positioning (other than throttled position)
5. Locked Valves
6. Removal of a TCCP (Temporary Configuration Change Package)
7. Verification of calculations
8. Welding related activities

4.3.2.3. For Clearance Application and Removal During Refueling, the following apply:

1. Independent Verifications (IV's) are performed for safety related equipment when the equipment's function is required in the current mode of operation. If the safety related equipment is not required in the current mode, IV and/or CV are not required for removal of equipment from service.
2. IV's shall be performed as required to support operability and prior to establishing conditions requiring the safety related equipments function.

4.3.2.4. Independent Verification involves the following process:

1. The performer of the component manipulation is separated from the verifier by time.
2. The performer shall, with use of the controlling document:
   - **LOCATE** the component and identify each unique identifier on the component label.
   - **PERFORM** the intended action.
3. The verifier shall, with use of the controlling document:
   - **LOCATE** the component and identify each unique identifier on the component label.
   - **CONFIRM** the completed action.

4.3.2.5 Independent Verification for calculations is performed by a second qualified individual using the same, or an authorized alternate methodology and documentation as the first individual. **(CM–4)**

4.3.3. Concurrent Verifications

4.3.3.1 Concurrent Verification Application

Concurrent Verification (CV) may be applicable to -but not limited to- component manipulations, clearance application / removal, performance of a procedure or other activities that removes equipment from service. Examples are:

1. Fuse removal and replacement
2. Lifting and re-landing leads
3. Booting relay contacts
4. Jumper installation
5. Valve throttling
6. Breaker manipulation
7. Switch manipulation
8. Gagging of valves

4.3.3.2. Examples of an irrecoverable condition with immediate consequences to the plant **or** threats to safe and continuous plant operation may include, but are not limited to:

1. Plant Scram
2. ESF Actuations
3. Reactivity Events
4. Unplanned Half Scrams
5. Unplanned containment partial or full isolations

4.3.3.3 Concurrent Verification involves the following process:

1. Both individuals involved determine, prior to the verification, who will fulfill the role of the performer the component manipulations and who will be the verifier. The individuals must rigorously adhere to these roles during concurrent verification.

Form ES-401-5 **Written Exam Question Worksheet** Q # 067

Examination Outline Cross-Reference Tier # 3

Evolution/System Conduct of Operations Group #

K/A # 2.1.8 Page # 2-1 RO/SRO Importance Rating 3.8 3.6

Measurement Ability to coordinate personnel activities outside the control room.

10CFR55.45.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Fire in the control room has caused a control room evacuation.
- Operating crew is implementing OP-TM-EOP-020, Cooldown From Outside Of Control Room.

From the list of in-plant actions below identify the ONE selection that is the HIGHEST PRIORITY (FIRST from the list to be performed).

A. Open both BWST suction valves MU-V-14A and MU-V-14B.

B. Trip both Main Feedwater Pumps.

C. Trip all Reactor Coolant Pumps.

D. Close RC-V-2, PORV isolation.

Technical Reference OP-TM-EOP-020, Cooldown From Outside Control Room, Section 2.0, Page 1, Rev. 4.

Open Exam Reference None.

Learning Objective V.D.18.02

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. Trip both Main Feedwater Pumps is the highest priority (step 2.4) Numbered steps are pre-prioritized to be performed in numerical order.

Distracter is plausible because this is a valid immediate action (step 2.6).

B CORRECT. Trip both Main Feedwater Pumps is the highest priority (step 2.4) Numbered steps are pre-prioritized to be performed in numerical order.

C INCORRECT. Trip both Main Feedwater Pumps is the highest priority (step 2.4) Numbered steps are pre-prioritized to be performed in numerical order.

Distracter is plausible because this is a valid immediate action (step 2.5).

D INCORRECT. Trip both Main Feedwater Pumps is the highest priority (step 2.4) Numbered steps are pre-prioritized to be performed in numerical order.

Distracter is plausible because this is a valid immediate action (step 2.7).

Comments None.

COOLDOWN FROM OUTSIDE OF CONTROL ROOM

1. **ENTRY CONDITIONS** - Fire in the relay room or Control Room, or another hazard which threatens to make the Control Room uninhabitable or threatens the ability to achieve safe shutdown from the Control Room.

2. **IMMEDIATE ACTIONS**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 2.1 **TRIP** the reactor	
___ 2.2 **PERFORM** OP-TM-EOP-001 Immediate Manual Actions.	**GO TO** Attachment 12.
___ 2.3 **INITIATE** OP-TM-EOP-010 Guide 15 "EFW Actuation Response".	**GO TO** Attachment 12.
___ 2.4 **TRIP** both main Feedwater Pumps.	**GO TO** Attachment 12.
___ 2.5 **TRIP** all reactor coolant pumps.	**GO TO** Attachment 12.
___ 2.6 **OPEN** MU-V-14A **and** MU-V-14B.	
___ 2.7 **CLOSE** RC-V-2.	

3. **FOLLOW-UP ACTIONS**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.1 Stabilize at Hot Shutdown	
___ 3.1.1 **ANNOUNCE** "reactor trip" and "______ requires commencing remote shutdown sequence", over the plant page and radio.	___ **PERFORM** Step 3.1.1 after Step 3.1.2 (communications isolated from control & relay rooms)

NOTE:

(1) The typical assignments for a cooldown outside of the CR are as follows. SM, STA & URO go to the control tower 2nd floor and then to the RSD panels, US & ARO go to the control tower 3rd floor and then to the RSD panels, Secondary reading AO goes to the EFW area, and Primary AO goes to MU valve alley.

(2) M&I phones should be utilized for communications between all of these locations. Cross tie primary and secondary M&I channels.

(3) Portable emergency lighting is available at RSD area, Maintenance Shop East of Roll Up Door, RB hatch and Primary AO Central

Form ES-401-5 **Written Exam Question Worksheet** Q # 068

Examination Outline Cross-Reference Tier # 3

Evolution/System Conduct of Operations Group #

K/A # 2.1.31 Page # 2-4 RO/SRO Importance Rating

Measurement Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup.

10CFR55.41(b)(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

10 CFR Part 55 Content ☐ 55.41 .6 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor is operating at 100% power.
- ICS Stations in MANUAL:
  - Reactor Master.
  - Diamond Rod Control.
  - Loop A and Loop B FW Loop Masters.
  - SG A/B Load Ratio.
  - Steam Generator Reactor Demand.

Based on these conditions, identify the ONE selection below that describes a condition the operator is required ENSURE prior to returning the Diamond Rod Control panel to automatic.

A. FAULT RESET pushbutton lamp LIT.

B. TR CF lamp LIT at the Manual Transfer pushbutton.

C. RUN SPEED selected at the Speed Selector switch.

D. SEQUENCE mode of operation selected at the Sequence/Sequence Override pushbutton.

Technical Reference OP-TM-621-471, ICS Manual Control, Step 5.2.2, Page 4, Rev. 0.

Open Exam Reference None.

Learning Objective GLO-10-624

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # QR-GLO-10-624-Q07

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the fault reset pushbutton is not required to be lit to transfer the diamond to automatic IAW OP-TM-621-471.

Distracter is plausible because the Fault Reset pushbutton is required to be pushed to clear a Motor Fault problem.

B INCORRECT because the TR CF lamp is not required to be lit to transfer the Diamond to automatic IAW OP-TM-621-471.

Distracter is plausible because the TR CF lamp would be lit on a transfer to or from the auxiliary power supply and it is located on the manual transfer pushbutton which could be misconstrued as necessary to take the diamond in and out of manual control.

C INCORRECT because Run Speed is not required to be selected to transfer the Diamond to Auto.

Distracter is plausible because Run Speed needs to be selected at the conclusion of transferring a control rod to or from the Auxiliary power supply.

D CORRECT. Required action IAW OP-TM-621-471.

Comments None.

5.0 RETURN TO NORMAL

5.1 **When** manual ICS control is no longer required, **then OBTAIN** CRS concurrence to place ICS to Auto.

5.2 Return Diamond Panel to Auto as follows:

5.2.1 **VERIFY** AUTO INHIBIT lamp Off.

5.2.2 **ENSURE** SEQ selected on SEQ/OR switch.

5.2.3 **VERIFY** neutron error 0%.

5.2.4 **VERIFY** NI power > 5% on selected channel (CC).

5.2.5 **OBTAIN** CRS concurrence to place Diamond Panel to Auto.

5.2.6 **PRESS** AUTO/MAN pushbutton on Diamond Panel.

1. **VERIFY** AUTO lamp Lit.
2. **ENSURE** NO control rod insertion or withdrawal occurs.

5.3 Return REACTOR DEMAND station to AUTO as follows:

5.3.1 **VERIFY** FW Valves in AUTO (FW-V-16A, FW-V-16B, FW-V-17A and FW-V-17B).

5.3.2 **PLACE** SG A FW DEMAND station indicator to HAND MINUS AUTO position.

5.3.3 **PLACE** SG B FW DEMAND station indicator to HAND MINUS AUTO position.

5.3.4 **ADJUST** SG/REACTOR DEMAND **and/or** SG A/B LOAD RATIO (ΔT_C) demand to obtain zero error on both A & B SG FW DEMAND station indicators.

1. **VERIFY** SG A FW DEMAND station indicator at 50%.
2. **VERIFY** SG B FW DEMAND station indicator at 50%.

Form ES-401-5 **Written Exam Question Worksheet** Q # 069

Examination Outline Cross-Reference Tier # 3

Evolution/System Equipment Control Group #

K/A # 2.2.12 Page # 2-6 RO/SRO Importance Rating 3.0 3.4

Measurement Knowledge of surveillance procedures.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** B.

Identify the ONE condition below that PROHIBITS the use of "N/A" for an in service test (IST) procedure step.

A. Performance of a partial test.

B. Skipping the step will result in an incomplete test.

C. Skipping the step will NOT result in missed acceptance criteria.

D. Precaution/Limitation/Prerequisite is NOT met - NO apparent impact on acceptance criteria.

Technical Reference HU-AA-104-101, Section 4.7.1.4, Page 6, Rev. 0.

Open Exam Reference None.

Learning Objective V.A.04.05

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. HU-AA-104-101 allows N/A for steps that are not being performed during a partial test.

Distracter is plausible because of the possible misconception that the procedure step is being used during the partial test.

B CORRECT. HU-AA-104-101 states that any step that may result in an incomplete surveillance cannot be skipped.

C INCORRECT. HU-AA-104-101 allows skipping steps that will not result in missed acceptance criteria.

Distracter is plausible because of the possible misconception procedure steps within a partial test sequence must be performed even if not relevant to the partial procedure.

D INCORRECT. HU-AA-104-101 allows N/A for prerequisites/limitations/precautions that are not applicable during the test.

Distracter is plausible because of the possible misconception that ALL precautions/limitations/prerequisites must be met at all times during the performance of a procedure.

Comments None.

4.7. Partial Performance

4.7.1. **If** a portion of a procedure is used in lieu of performing the procedure in its entirety **then** the job supervisor of the individual performing the procedure or work planner will:

SQR Stations

4.7.1. **If** a portion of a procedure is used in lieu of performing the procedure in its entirety and the procedure is written to allow this, **then** the appropriate person having the authority stated in the procedure can authorize partial use. **If** no partial procedure use authority is stated, **then** the Station Qualified Reviewer (SQR) will:

1. **DETERMINE** the steps that are adequate and appropriate to accomplish the desired task.

2. **ENSURE** all applicable Prerequisites, Precautions, and Limitations and Actions are met before performing.

3. **ENSURE** the component/system is returned to a condition ready to perform the next evolution or returned to condition normal/expected for plant conditions at that time.

4. **ENSURE** that skipping steps will not result in missed acceptance criteria or an incomplete surveillance.

5. **INDICATE** that the procedure is partially performed and why.

    A. **ANNOTATE** steps that are not applicable before performing a partial procedure with "N/A".

        1. Steps that may be required during the course of the work may be annotated when the job has been completed.

4.8. Remote Performance

4.8.1. **When** a secondary individual(s) is required to perform a portion of a procedure, **then** they shall have at a minimum:

- A copy of the applicable steps to permit placekeeping for their portion of the procedure, and
- Any pertinent Precautions or Limitations and Actions.

4.8.2. **If** a procedure requires remote actions by a secondary individual(s), **then** the in-hand and place keeping requirements may be satisfied by the primary individual through formal communication of the required actions to the secondary individual(s).

Form ES-401-5 **Written Exam Question Worksheet** Q # 070

Examination Outline Cross-Reference Tier # 3

Evolution/System Equipment Control Group #

K/A # 2.2.25 Page # 2-7 RO/SRO Importance Rating 2.5 3.7

Measurement Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor operating at 80% power with ICS in automatic.
- Nuclear engineers collecting NAS data regarding core power distribution.
- Control Rod Index 250.
- Reactor power imbalance NEGATIVE 32%.

Event:
- Uncontrolled rod withdrawal begins.

Based on these conditions identify the ONE selection that describes the FIRST reactor trip setpoint to be exceeded, and the basis for that trip.

A. Nuclear overpower, to protect fuel integrity.

B. High RCS pressure, to maintain RCS pressure boundary integrity.

C. High RCS outlet temperature, to limit coolant temperature during reactor operation.

D. Nuclear overpower based on RCS flow and power imbalance, to maintain DNBR greater than minimum requirements.

Technical Reference TMI Technical Specification Section 2.2, Page 2-4, Amendment 157.
TMI Technical Specification Page 2-7, Amendment 247.

Open Exam Reference None.

Learning Objective IV.E.14.03

Question Source ☑ New ☐ Bank ☐ Modified Bank Question # Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because from 80% power, other trip setpoints will be reached prior to 105.1%.

Distracter is plausible because reactor power will rise due to the control rod outmotion.

B CORRECT. RCS pressure will rise rapidly due to RCS Tave rise causing an insurge into the pressurizer, compressing the steam bubble.

C INCORRECT because other trip setpoints will be reached prior to RCS Thot reaching 618.8 degrees F.

Distracter is plausible because RCS Thot will rise due to the control rod outmotion and rising reactor power.

D INCORRECT because as rods withdraw, the power imbalance will become less negative, not more negative, therefore this trip setpoint will not be reached.

Distracter is plausible because with the large power inbalance in the stem, if inbalance was to get more negative, then the reactor would trip on high flux/flow/inbalance.

Comments None.

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system (Reference 1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure (Reference 2). The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established (Reference 2). The maximum settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by the presence of a pressurizer electromatic relief valve (Reference 3).

References

(1) UFSAR, Section 4.0 - "Reactor Coolant System"

(2) UFSAR, Section 4.3.10 - "Safety Limits and Conditions"

(3) UFSAR, Table 4.2-8 - "Reactor Coolant System Pressure Settings"

the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced.

b. Pump Monitors

The redundant pump monitors prevent the minimum core DNBR from decreasing below the Statistical Design Limit of 1.313 (BWC) by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure ensures that the system pressure is maintained below the safety limit (2750 psig) for any design transient (Reference 2). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

As part of the post-TMI-2 accident modifications, the high pressure trip setpoint was lowered from 2390 psig to 2300 psig. (The FSAR Accident Analysis Section still uses the 2390 psig high pressure trip setpoint.) The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

A B&W analysis completed in September of 1985 concluded that the high reactor coolant system pressure trip setpoint could be raised to 2355 psig with negligible impact on the frequency of opening of the PORV during anticipated over-pressurization transients (Reference 3). The high pressure trip setpoint was subsequently raised to 2355 psig. The potential safety benefit of this action is a reduction in the frequency of reactor trips.

The low pressure and variable low pressure trip setpoint were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (References 4, 5, and 6). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. The revised low pressure trip of 1900 psig and the variable low pressure ($16.25\ T_{out}$ - 8113) trip setpoint prevent the minimum core DNBR from decreasing below the Statistical Design Limit of 1.313 (BWC). Figure 2.3-1 shows the high pressure, low pressure, high temperature and variable low pressure trip setpoints.

Form ES-401-5 **Written Exam Question Worksheet** Q # 071

Examination Outline Cross-Reference Tier # 3

Evolution/System Radiation Control Group #

K/A # 2.3.11 Page # 2-10 RO/SRO Importance Rating 2.7 3.2

Measurement Ability to control radiation releases.

10CFR55.41(b)(12) Radiological safety principles and procedures.

10 CFR Part 55 Content ☑ 55.41 .12 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** C.

Plant conditions:
- OTSG 1A tube leak (300 gpm) shutdown in progress using OP-TM-EOP-005.
- Reactor was manually tripped 25 minutes ago.
- RCS T-Hot is 510 degrees F.
- RCS pressure is 980 psig.
- OTSG 1A/1B levels are both 25 inches.
- BWST level is 24 feet.
- Tech Support Center (TSC) is NOT staffed at this time.

The Radiological Assessment Coordinator (RAC) reports that projected off-site doses based on an 8-hour duration release are:
- 360 mrem Whole Body.
- 2160 mrem Child Thyroid.

Based on these conditions, identify the ONE statement below that describes required OTSG 1A isolation status, AND the basis for that status.

A. Must NOT be isolated due to BWST level.

B. Isolation is required due to high tube leak rate.

C. Isolation is required due to high dose projections.

D. Must NOT be isolated due to RCS temperature and pressure.

Technical Reference OP-TM-EOP-005, OTSG Tube Leakage, Step 3.16, Page 7.

Open Exam Reference None.

Learning Objective V.E.05.09

Question Source ☐ New ☑ Bank Question # NRC 20 Q-096.

☐ Modified Bank Parent Question #

Question NRC Exam History TMI 2001 Q-096

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the BWST level is not at a level requiring operator action.

Distracter is plausible because low BWST level is a valid reason to isolate the OTSG and BWST level higher than that could be interpreted as a reason not to isolate the OTSG.

B INCORRECT because the tube leak rate is not high enough to require OTSG isolation on high OTSG level.

Distracter is plausible because if the leak rate is too large, then OTSG isolation will be required when OTSG level is > 85%.

C CORRECT. Child Thyroid dose rate is > 250 mrem/hr, the OTSG isolation criteria.

D INCORRECT because RCS pressure is less than 1000 psig.

Distracter is plausible because RCS pressure too high is a valid reason to not isolate the OTSG due to the possibility of lifting the Main Steam Safety Valves.

Comments None.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3.16 **IAAT** any of the following conditions exists (without contrary TSC guidance): _____ OTSG level > 85% Operate Range _____ BWST level < 21 ft _____ Projected **or** actual offsite dose rates approach 50 mrem/hr W.B. (TEDE) **or** 250 mrem/hr child thyroid (CDE), **when** RCS pressure < 1000 psig, **then** _____ **INITIATE** Attachment 1A or B to isolate the affected OTSG(s). **When** affected OTSG TBV/ADVs are closed, **then** _____ **If** both OTSGs are being isolated, **then** **GO TO** OP-TM-EOP-009. _____ **PERFORM** Guide 12 "RCS Stabilization Following OTSG Isolation".	
NOTE If a pressurizer cooldown rate above 100°F in an hour or a RCS cooldown rate above 100°F/HR with RCS below 525°F is needed to permit isolation of the OTSG, then invoke 10CFR50.54x.	
☐ 3.17 **IAAT** OTSG isolation criteria may be challenged prior to reducing RCS pressure < 1000 psig, **then** _____ 1. **INITIATE** RCS cooldown to 500°F at a rate within RCS inventory control capability **and** < 240°F/HR. _____ 2. **ENSURE** RC-V-2 is OPEN _____ 3. **CYCLE** the PORV to reduce SCM to approximately 30 °F.	
_____ 3.18 **INITIATE** Guide 8 to MINIMIZE SCM.	

Form ES-401-5 **Written Exam Question Worksheet** Q # 072

Examination Outline Cross-Reference Tier # 3

Evolution/System Radiation Control Group #

K/A # 2.3.9 Page # 2-9 RO/SRO Importance Rating 2.5 3.4

Measurement Knowledge of the process for performing a containment purge.

10CFR55.41(b)(12) Radiological safety principles and procedures.

10 CFR Part 55 Content ☑ 55.41 .12 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Plant cooldown in progress, preparing to establish Decay Heat Removal core cooling.
- Waste Gas Release Permit has been approved for an RB purge in accordance with OP-TM-823-406, RB Purge - Containment Closed.

Based on these conditions identify the ONE selection that describes the process for initiating this Reactor Building purge.

A. (1) Set RB Purge Manual Loader to desired value.
(2) Start Purge Supply Fans.
(3) Start Purge Exhaust Fans.

B. (1) Start Purge Supply Fans.
(2) Start Purge Exhaust Fans.
(3) Adjust RB Purge Manual Loader to desired value.

C. (1) Set RB Purge Manual Loader to desired value.
(2) Start Purge Exhaust Fans.
(3) Start Purge Supply Fans.

D. (1) Start Purge Exhaust Fans.
(2) Start Purge Supply Fans.
(3) Adjust RB Purge Manual Loader to desired value.

Technical Reference OP-TM-823-406, RB Purge - Containment Closed, Pages 4 - 6, Rev. 0.

Open Exam Reference None.

Learning Objective None.

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # Robinson 2004 NRC Exam

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the sequence of events is incorrectly listed.

Distracter is plausible because the correct events are listed.

B INCORRECT because the sequence of events is incorrectly listed.

Distracter is plausible because the correct events are listed.

C INCORRECT because the sequence of events is incorrectly listed.

Distracter is plausible because the correct events are listed.

D CORRECT. The correct events are listed in the correct order.

Comments Modified Robinson 2004 NRC Exam Question.

4.0 MAIN BODY

4.1 **SET** AH-FT-148B or AH-FT-909 alarm potentiometer to value specified on Waste Gas Release Permit using Attachment 7.2 or 7.3, as applicable. ____

4.2 **SELECT** AH-FT-148B or AH-FT-909 for alarm input on H&V Panel as applicable. ____

NOTE: Fan suction damper will open allowing fan to start when fully open.

4.3 **START** the following:

- AH-E-7A ____
- AH-E-7B ____

4.4 **OPEN** AH-V-1A.

4.5 **PLACE** lock key switch for AH-V-1B in OPENING PERMITTED. ____

4.6 **OPEN** AH-V-1B. ____

4.7 **PLACE** lock key switch for AH-V-1B in OPENING DEFEATED. ____

NOTE: Purge valves can only be opened to 90° if limits are removed and unit is in cold shutdown or refueling shutdown.

4.8 **If** purge valves can be fully opened (90°),
then perform the following:

4.8.1 **OPEN** AH-V-1B-BK, 1A ES VALVES MCC Unit 1B. ____

4.8.2 **HAND CRANK** AH-V-1B to the full open 90° position. ____

4.8.3 **CLOSE** AH-V-1B-BK, 1A ES VALVES MCC Unit 1B. ____

NOTE: Fan discharge damper will open allowing fan to start when fully open.

4.9 **If** max allowable purge rate is ≤ 25,000 scfm,
and purge valves can be fully opened (90°),
then START <u>one</u> of the following:

- AH-E-6A ____
- AH-E-6B ____

4.10 **If** max allowable purge rate is > 25,000 scfm,
or purge valves are limited to 30° open,
then START both of the following:

- AH-E-6A ____
- AH-E-6B ____

4.11 **OPEN** AH-V-1D. ____

4.12 **PLACE** lock key switch for AH-V-1C in OPENING PERMITTED. ____

4.13 **OPEN** AH-V-1C. ____

4.14 **PLACE** lock key switch for AH-V-1C in OPENING DEFEATED. ____

NOTE: Purge valves can only be opened to 90° if limits are removed and unit is in cold shutdown or refueling shutdown.

4.15 **If** purge valves can be fully opened (90°),
then perform the following:

4.15.1 **OPEN** AH-V-1C-BK, 1B ES VALVES MCC Unit 1B. ____

4.15.2 **HAND CRANK** AH-V-1C to the full open 90° position. ____

4.15.3 **CLOSE** AH-V-1C-BK, 1B ES VALVES MCC Unit 1B. ____

NOTE: Normally desired purge rate is the maximum allowable purge rate specified on the Waste Gas Release Permit. The SM/CRS may prescribe a lesser flow rate if desired. If purge valves are limited to 30° open, maximum flow rate achievable 14,000 scfm. Purge supply fan discharge temperature should be within limits within 5 minutes of starting purge. A reduction in purge flow will raise purge supply temperature.

4.16 **ADJUST** RB Purge Manual Loader Purge Rate (AH-D-8B-EX1) to achieve the following:

- Desired purge rate <u>not to exceed</u> maximum allowable purge rate as indicated on AH-FR-909 (blue pen on FR-148 for purge rates < 20,000 scfm) or AH-FR-148B (green pen on FR-148 for purge rates ≥ 20,000 scfm). ____
- Purge supply fan discharge temperature ≥ 90°F (containment integrity required) or ≥ 55°F (containment integrity **not** required) as locally indicated by AH-TI-6A and/or AH-TI-6B. ____

4.17 **MARK** FR-148 with start time, date, and release number. ____

4.18 **MONITOR** purge supply fan discharge temperature (AH-TI-6A and/or AH-TI-6B) at least once/shift. ____

4.19 **IAAT** either of the following conditions exist:

- Purge supply fan discharge temperature **cannot** be maintained ≥ 90°F with containment integrity required
- Purge supply fan discharge temperature **cannot** be maintained ≥ 55°F with containment integrity **not** required

then GO TO Section 5.0 to stop RB purge. ☐

4.20 **INDICATE** purge rate obtained on Waste Gas Release Permit. ____

4.21 **If** restarting RB purge from a temporary shutdown,
then COMPLETE release restart time information and reason for stop/restart on Waste Gas Release Permit. ____

4.22 **MAINTAIN** purge information as required on Waste Gas Release Permit. ____

Form ES-401-5 **Written Exam Question Worksheet** Q # 073

Examination Outline Cross-Reference Tier # 3

Evolution/System Emergency Procedures/Plan Group #

K/A # 2.4.12 Page # 2-12 RO/SRO Importance Rating 3.4 3.9

Measurement Knowledge of general operating crew responsibilities during emergency operations.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Identify the ONE selection below that describes procedure place keeping requirements during implementation of EMERGENCY OPERATING PROCEDURES (EOPs).

During EOP implementation ____________________ are REQUIRED to be checked or otherwise marked.

A. only transitions between procedures

B. only steps provided with check-off spaces

C. all EOP steps, whether or not check-off spaces are provided

D. all steps with check-off spaces, and ALL EOP Rule/Guide steps

Technical Reference OS-24, Conduct of Operations During Abnormal and Emergency Events, section 4.1.14, Page 14, Rev. 10.
TMI Operations Expectation Database, Placekeeping SOS Response dated 10/06/2004.

Open Exam Reference None.

Learning Objective LP 11.2.01.513, Obj. 2.

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because transition between procedures is not the only time procedures are required to be marked for place keeping.

Distracter is plausible because transition between procedures requires an announcement by the CRS and could be interpreted as the only time for procedure placekeeping.

B INCORRECT because most EOP-010 rules and guides do not have check-off spaces but are required to be marked for place keeping.

Distracter is plausible because steps provided with check-off spaces are required to have those spaces marked as the step is performed.

C INCORRECT because it does not address EOP-010 rules guides and graphs.

Distracter is plausible because place keeping in EOPs is an extremely important operator fundamental.

D CORRECT. Operations expectations are that all rules and guides steps will be checked for place keeping purposes.

Comments None.

4.1.14 Place keeping in an EVENT PROCEDURE

A. Check-off spaces are checked or otherwise marked after the action required by the step is completed. If the procedure is re-performed, additional marks are used.

B. Check-off spaces for VERIFY steps when used in two column format, are completed as follows. If the condition is satisfied, mark the space for the VERIFY step and leave the right hand column spaces blank. If the condition is not satisfied, leave the VERIFY space blank, and mark the spaces in the right hand column after the action required by the step is complete.

C. 24 Hour clock time should be entered in the TIME spaces which occur periodically throughout the EOP. These reference times are used to perform time dependent actions or to reconstruct the event.

D. EOP Rules posted on the Control Boards contain check-off spaces that are not required to be checked or otherwise marked as the step is performed by Reactor Operators. The check-off spaces are marked afterward as a verification that the Rule was performed correctly.

E. CARRYOVER STEPS are left blank until the step applies, and marked NA after the procedure is completed if the step condition was not satisfied.

4.1.15 TWO COLUMN Format

A. The user of the procedure reads the "ACTION/EXPECTED RESPONSE" from the left hand column.

B. If the action is completed satisfactorily or if the response is as expected, then the user proceeds down to the next step in the left hand column (and skips the right hand "Response not obtained" column)

C. If the action cannot be completed or the response is not as expected, then the user proceeds to the right hand column. The user takes the action described in the right hand column and proceeds to the next step in the left hand column.

D. If a "**VERIFY**" step is used in the LH column and no RNO is specified, then do not proceed past this step if the condition is not satisfied.

TMI Operations Expectation

The objective of the expectations process is to assure uniform application of expectations and standards in the plant and the training environment. The expectations process document provides a method for instructors, students and operations license holders to solicit clarification from the SOS when clarification from the employees immediate management does not resolve the issue. In the case of training instructors it is a direct method to obtain documented feedback on Operations Expectations and Standards. It can be a method to determine how to implement certain aspects of procedures but it is not a method to change procedures. Deficient procedures are addressed through IR process and enhancements are through direct discussion with the procedure owner.

Date: 10/06/2004

Originator: Ken McCall/TMI Extention: 2061

Description: Placekeeping is not consistently executed when ROs utilize the hardcards for Rules & Guides. Some provide checkoff lines and others do not. Some operators execute the guide then placekeep when verifying their actions.

Fundamental: Procedure Adherence Procedure:

Recomendation: Placekeeping is performed on all Rules & Guides as they are performed.

Date: 10/06/2004

Response by SOS: Placekeeping shall be performed on all Rules & Guides as they are performed if there are no signoff lines the user should placekeep as well.

Feedback Mechanism:

Reply to:

- [x] TMI_SRO
- [ ] TMI_CRO
- [ ] TMI_AO
- [x] TMI_Training Ops Group

Form ES-401-5 **Written Exam Question Worksheet** Q # 074

Examination Outline Cross-Reference Tier # 3

Evolution/System Emergency Procedures/Plan Group #

K/A # 2.4.2 Page # 2-11 RO/SRO Importance Rating 3.9

Measurement Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

10CFR55.41(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** B.

Initial plant conditions:
- Reactor operating at 100% power with ICS in full automatic.

Current plant conditions:
- Unexpected FW flow reduction to 8E6 lbm/hr occurred.
- Pressurizer level is now at 280 inches, rising at 5-inches per minute.
- Reactor coolant outlet temperature is now 619 degrees F, rising at 2 degrees per minute.
- RC-V-1 Pressurizer Spray Valve has operated automatically as designed.
- RCS pressure is now 2300 psig, rising at 10 psig per minute.

Identify the ONE statement below that describes the immediate operator action required for this condition.

A. Initiate plant shutdown.

B. Manually trip the reactor.

C. Manually reduce reactor power to correspond to existing FW flow.

D. Manually operate RCS letdown and makeup controls to lower Pressurizer level to 220 inches.

Technical Reference OS-24, Conduct of operations During Abnormal and Emergency Events, Attachment A , Licensed Operator Memory Items, Page 32, Rev. 10.

Open Exam Reference None.

Learning Objective V.E.13.01

Question Source ☐ New ☑ Bank Question # QR5E13-01-Q04

☐ Modified Bank Parent Question # V.E.13.01

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT. Initiation of plant shutdown would reduce reactor power. However, since the RPS has failed to actuate an automatic trip, EOP-001 and OS-24 procedures require the reactor to be manually tripped immediately.

Distracter is plausible because shutting down the plant will prevent the reactor from exceeding the design safety analysis setpoint of 620 degrees F.

B CORRECT. Immediate manual reactor trip is correct. EOP-001 entry conditions require the reactor to be manually tripped immediately for any unplanned condition requiring an automatic or manual trip signal. OS-24 procedure states that a reactor trip is required if parametric limits are exceeded, specifically including RCS Thot greater than 618 degrees F.

C Manual reactor power reduction would reduce the power mismatch. However, since the RPS has failed to actuate an automatic trip, EOP-001 and OS-24 procedures require the reactor to be manually tripped immediately.

Distracter is plausible because reducing reactor power will reduce RCS temperature, pressure and pressurizer level, countering the rise caused by the reduced feedwater flow.

D Operation of RCS letdown and makeup controls could impact the rising Pressurizer level. However, since the RPS has failed to actuate an automatic trip, EOP-001 and OS-24 procedures require the reactor to be manually tripped immediately.

Distracter is plausible because lowering pressurizer level will cause RCS pressure to reduce, lowering pressure away from it's trip setpoint of 2350 psig.

Comments None.

2. Reactor Trip Requirements:

2.1 A reactor trip is required (automatic or manual) if any of the following limits are exceeded:

- Reactor power is > 105.1%
- RCS Thot is > 618 °F
- RCS pressure > 2350 psig
- RCS pressure < 1900 psig
- Containment pressure > 4 psig
- Reactor Power >55% with less than 3 RCPs operating
- No RCP operating in one loop
- Reactor power above flux/flow/axial imbalance limit (Limit is not a memory item. The required action is memory item. SPDS or COLR figure is used to determine if limit is exceeded)
- Turbine trip and >45% reactor power
- Both A and B Main Feedwater pump turbines trip and > 7 % reactor power.

2.2 **If** a reactor trip is required by any event procedure, **then** PRESS both REACTOR TRIP **and** DSS pushbuttons.

Form ES-401-5 **Written Exam Question Worksheet** Q # 075

Examination Outline Cross-Reference Tier # 3

Evolution/System Emergency Procedures/Plan Group #

K/A # 2.4.1 Page # 2-11 RO/SRO Importance Rating 4.3 4.6

Measurement Knowledge of EOP entry conditions and immediate action steps.

10CFR55.41(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D.

Initial conditions:
- Reactor operating at 100% power with ICS in full automatic.

Event:
- Circulating Water tube leak on B side of the Main Condenser.

Current conditions:
- CE-6A (corrected feedwater cation conductivity) control room indication is pegged high at 3.0 micromho/cm.
- Chemistry has validated the following sample readings:
  - CE-6A (corrected feedwater cation conductivity) is 6.5 micromho/cm
  - CE-6 (feedwater cation conductivity) is 7.5 micromho/cm

Based on these conditions identify the ONE statement below that describes required actions.

A. Secure all Moisture Separator Drain Pumps and continue power operation.

B. Reduce power to less that 50% and isolate the "B" side Circulating Water loop.

C. Perform a normal plant shutdown and cooldown to Decay Heat Operations.

D. Trip the reactor and go to OP-TM-EOP-001, Reactor Trip.

Technical Reference 1203-5, High Contam9inanats in the Condensate and/or Feedwater System, Immediate Manual Action 2.B.2, Page 2, Rev. 26.

Open Exam Reference None.

Learning Objective V.C.02.03

Question Source ☐ New ☑ Bank Question # SR5C02-03-Q02.

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because 1203-5 requires reactor to be tripped under these conditions.

Distracter is plausible because this action would stop recirculation of the contaminants.

B INCORRECT because 1203-5 requires reactor to be tripped under these conditions.

Distracter is plausible because this is the action required for lower contamination levels.

C INCORRECT because 1203-5 requires reactor to be tripped under these conditions.

Distracter is plausible if the examinee does not realize the immediacy of the problem.

D CORRECT answer - in accordance with 1203-5 if CE-6A >5.0 micromhos/cm or CE-6 > 6.0 micromhos/cm.

Comments NRC CRO Licensing Examination June 2000.

1.0 **SYMPTOMS**

1. Increasing conductivity on Control Room conductivity recorder.
2. Increasing conductivity on secondary sampling recorders.
3. Increasing sodium on the sodium monitor at the condensate pump discharge.
4. Alarm PLB-8-6, "Conductivity Recorder Abnormal".
5. Alarm PRF 6-1, "CE-6A Conductivity Hi".
6. Alarm PRF 2-6, "CO-C1-A Conductivity Trouble".
7. Alarm PRF 2-7, "CO-C1-B Cndtvty Trouble".
8. Alarm PRF 2-8, "CO-C2A/B Cndtvty Trouble".
9. Alarm PLB-5-7, "Turbine Sampling Room Trouble", caused by increasing sodium in the OTSG Feedwater and in the Condensate System. (OTSG Feedwater over 3 ppb and increasing.)
10. Alarm on the following PPC (Plant Process Computer) points:
    1. A1031 - output from instrument CE-773 (normally monitors CE-3, Condensate Pump Discharge)
    2. A1032 - output from instrument CE-772 (normally monitors CE-2, Condensate Pump Outlet and also monitors Ecolochem Makeup Water)
    3. A1033 - output from instrument CE-801 (normally monitors CE-6, OTSG Feedwater)

2.0 **IMMEDIATE ACTION**

A. Automatic - None

B. Manual

1. **CONTACT** Chemistry Dept. to confirm abnormal conductivity with the cation conductivity recorders in the secondary sample room.
2. **IF** CE-6A (corrected feedwater cation conductivity) is confirmed > 5.0 μmho/cm or CE-6 (feedwater cation conductivity) is confirmed > 6.0μmho/cm, **THEN** immediately **TRIP** the reactor **AND GO TO** OP-TM-EOP-001, Reactor Trip.

Form ES-401-5 **Written Exam Question Worksheet** Q # 076

Examination Outline Cross-Reference Tier # 1

Evolution/System 025 Loss of RHR System Group # 1

K/A # 2.1.14 Page # 2-2 RO/SRO Importance Rating 2.5 3.3

Measurement Conduct of Operations: Knowledge of system status criteria which require the notification of plant personnel.
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** D.

Initial plant conditions:
- Maintenance outage in progress.
- Reactor vessel head is removed
- DHR Train "A" aligned for RCS cooling.
- Incore thermocouple temperatures steady at 100 degrees F.

Event:
- Decay Heat Removal Pump discharge pressure and DHR system flow rate begin to oscillate excessively.
- Incore thermocouple temperatures are now 115 degrees F, continuing to rise.

Based on these conditions identify the ONE selection below that describes required operator actions, and the appropriate procedure to be implemented.

A. Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.

B. Reduce DH Train A flow using OP-TM-212-451, Control of DH Train A Flow and temperatures, until the flow and pressure oscillations stop.

C. Place DH Train B in service IAW OP-TM-212-901, Emergency DHR Operations, and then vent DH-P-1A using OP-TM-212-553, Vent of DH-P-1A.

D. Stop DH-P-1A and evacuate all personnel from the Reactor Building (RB) by actuating the RB Evacuation alarm IAW EOP-030, Loss of Decay Heat Removal.

Technical Reference OP-TM-EOP-030, Loss of Decay Heat Removal, Step 3.3, Page 3, Rev. 0.

Open Exam Reference None.

Learning Objective V.D.16.04

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because the EOP-010 is not applicable while on Decay heat Removal cooling.

B INCORRECT answer because stem conditions exceed EOP-030 procedure entry threshold conditions, and the actions presented here do not comply with EOP-030 guidance.

C INCORRECT answer because stem conditions exceed EOP-030 procedure entry threshold conditions, and the actions presented here do not comply with EOP-030 guidance.

D CORRECT answer.

Comments None.

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE			RESPONSE NOT OBTAINED
___		TIME	
___	3.1	**STOP** any procedure in progress which is reducing RCS inventory	
___	3.2	**If** fuel handling operations are in progress in the reactor building**, then NOTIFY** fuel handling SRO to place fuel down in deep end of FTC with grapple engaged	
___	3.3	**ACTUATE** Reactor Bldg. Evacuation alarm **and ANNOUNCE** "Decay Heat Cooling has been interrupted, ALL personnel shall exit the reactor building."	
___	3.4	**REQUEST** SM to evaluate Emergency Action Levels (EALs).	
___	3.5	**VERIFY** 1D **or** 1E 4160V Bus is energized.	_____ **INITIATE** OP-TM-861-901 (EG-Y-1A) **or** OP-TM-861-902 (EG-Y-1B) for the affected bus. _____ **If** EG-Y-1A **or** EG-Y-1B failed to load**, then INITIATE** OP-TM-864-901 to energize 1D or 1E 4160V bus from EG-Y-4
☐	3.6	**IAAT** at least one train of DHR is operating **and** incore temperature < 140°F **and <u>not</u>** rising, **then GO TO** section 6.0 "Return to Normal"	
☐	3.7	**IAAT** no DHR trains are available **and** incore temperature is <u>not</u> being controlled by OTSG heat removal, **then** ____ 1. **ENSURE** all reactor coolant pumps are SHUTDOWN. ____ 2. **If** the RCS is filled and pressure > 20 psig, **then GO TO** Section 4.0 "Emergency OTSG Cooling" ____ 3. **GO TO** Section 5.0 "Feed and bleed core cooling"	

Form ES-401-5 **Written Exam Question Worksheet** Q # 077

Examination Outline Cross-Reference

Tier # 1

Evolution/System 027 Pressurizer Pressure Control System (PZR PCS) Malf

Group # 1

K/A # AA2.05 Page # 4.2-21 RO/SRO Importance Rating 3.2 3.3

Measurement Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressurizer Heater Setpoints.
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- RC-1 LT1 and RC-2 TE1 are selected for inputs to the Pressurizer level recorder.
- RCS-Pressurizer boron equalization in progress using OP-TM-220-461, Equalize RCS and Pressurizer Boron Concentration.
  - Pressurizer Spray Valve RC-V-1 is partially open in manual.
  - Console Right selector switches for Pressurizer Heater Banks 1-5 turned to the ON POSITION.

Event:
- SUBFEEDS - AUTO/HAND lamp de-energizes (NOT lit).
- MAP G-3-5, Pzr level Lo-Lo actuates.
- Operator determines this alarm is invalid, based on redundant indications.

Based on these conditions identify the ONE selection below that describes:
(1) Lo-Lo Pressurizer level heater cut-out setpoint.
(2) Impact on Console Right Bank 1-5 heater control lamps.
(3) Applicable procedure.

A. (1) < 63.5 inches.
(2) Red lamps remain LIT due to manual control mode.
(3) 1202-41, Total or Partial Loss of ICS/NNI HAND Power.

B. (1) < 63.5 inches.
(2) Red lamps NOT lit.
(3) 1202-42, Total or Partial Loss of ICS/NNI AUTO Power.

C. (1) < 80 inches.
(2) Red lamps remain LIT due to manual control mode.
(3) 1202-41, Total or Partial Loss of ICS/NNI HAND Power.

D. (1) < 80 inches.
(2) Red lamps NOT lit.
(3) 1202-42, Total or Partial Loss of ICS/NNI AUTO Power.

Technical Reference 1202-42, Total or Partial Loss of ICS/NNI Auto Power, Steps 2.1.4 and 3.A.k, Pages 3 and 4, Rev. 42.

Open Exam Reference None.

Learning Objective V.D.22.04

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because setpoint for Pressurizer Heater Cut-out is at 80 inches, and the procedure identified is not correct.

Distracter is plausible because uncovery of the first (highest elevation) heater begins at 63.5 inches Pressurizer level, red lamp response is correct.

B INCORRECT answer because setpoint for Pressurizer Heater Cut-out is at 80 inches.

Distracter is plausible because it describes the correct red lamp response and identifies the correct procedure. Additional plausibility is merited because uncovery of the first (highest elevation) heater begins at 63.5 inches Pressurizer level.

C INCORRECT answer because the red lamp response and the procedure identified are both not correct.

Distracter is plausible because it contains the correct setpoint for the Pressurizer Heater Cut-out, possible misconception that manual heater operation in the stem ovverrides the Lo-Lo cut-out, and the procedure identified is correct.

D CORRECT answer.

Comments None.

2. Signals transfer to provide valid main feedwater flow indication (recorder) and valid feedwater valve ΔP indication.

3. MU-V-1A/B and MU-V-3 go closed due to auto powered temperature interlocks.

4. All pressurizer heater controls are "Off " due to pressurizer lo-lo level interlock.

5. MU-V-8 travels to "thru" position due to MU tank level interlock.

6. PORV (RC-RV-2) will not respond to automatic setpoints but is operable with the manual control switch.

7. Pressurizer Spray Valve (RC-V-1) will fail closed in Auto, but is operable in Manual mode.

8. MS-V-4A/B will transfer to back-up manual loader ("BACKUP CTRL" Bailey Stations)

2.2 Manual Action

NOTE

Control Room indications affected and alternate indications are listed on Table 1. Additionally, Table 3 provides indicators unaffected by the loss of auto power.

1. **VERIFY/ADJUST** plant control to obtain a stable plant configuration.

CAUTION

If 1. or 2. below cannot be performed as written (i.e., ATWS or failure of main turbine stop valves to close) go directly to OP-TM-EOP-001 at that point for direction on performance of remedial actions. Refer to this procedure for additional guidance.

2. **IF** feedwater control cannot be established, **THEN**: (NA this step if not required.)

    1. **TRIP** the reactor and **VERIFY** power less than **10%**.

    2. **TRIP** the main turbine and **VERIFY** T/G stop valves **CLOSED**.

    3. **TRIP** both main feedwater pumps.

    4. **GO TO** OP-TM-EOP-001 and refer to this procedure for additional guidance.

3.0 **FOLLOW-UP ACTIONS**

Objective: Stabilize the plant at power if the Reactor does not trip or at hot shutdown if the Reactor trips, and then restore "Auto" power.

> **NOTE**
>
> "A" is the priority step. Other steps can be performed while completing "A".

A. **CONTROL** the following components in manual as necessary: NA components not controlled in manual.

_____	a.	Reactor/Control Rods	Diamond Control Panel (CC)
_____	b.	Turbine/Control Valves	EHC OWS Panel (CL)
_____	c.	FW-V-16&17A/B	H/A Station (CC)
_____	d.	FW-P-1A&B	H/A Station (CL)
_____	e.	Maintain desired pressurizer level with HAND control of MU-V-17.	
_____	f.	MU-V-32	H/A Station (CC)
_____	g.	MS-V-3A-F	H/A Station (CC)
_____	h.	MS-V-4A/B	B/U Loader (CC)
_____	i.	RC-V-1	Place in manual then use open/close PB (CC)
_____	j.	RC-RV-2	Manual Control Switch (CC)

_____ k. Pzr Heaters

- _____ 1. **BYPASS** Lo Lo Level interlock with keyswitch (Key #2) in ICS/NNI pwr monitor cabinet (Key #214) if valid pzr level is ≥80" LI-777A.
- _____ 2. **OPERATE** Bank 4&5, using on-off switches (CR)
- _____ 3. **OPERATE** Banks 1, 2 and 3, from NNI Station (CC)

_____ l. Letdown Flow Control (MU-V-1A/B & MU-V-3)

- _____ 1. **DEFEAT** interlocks for MU-V-1A/1B by **LIFTING LEAD** on **7-3-3-16** in ICS/NNI Cabinet #7.
- _____ 2. **DEFEAT** interlock for MU-V-3 by **LIFTING LEAD** on **5-4-5-4** in ICS/NNI Cabinet #5.
- _____ 3. **REOPEN** MU-V-1A/B.
- _____ 4. **RESTORE** letdown per OP-TM-211-950. (Recovery from Letdown Line Isolation.)
- _____ 5. **RECORD** lifted leads per Enclosure 5 of AP 1013.

> **NOTE**
>
> If makeup Auto power is lost, there will be no letdown flow indication. MU-V-5 demand of 1% equals approximately 1 GPM.

_____ m. Letdown Bleed Control (MU-V-8)

- _____ 1. **VERIFY** MU Tk level is >18" on LI-778A.
- _____ 2. **DEFEAT** interlock by placing a **JUMPER** from **5-4-5-17 TO 5-4-5-18** in ICS/NNI Cabinet #5.
- _____ 3. **OPERATE** MU-V-8 as desired.
- _____ 4. **RECORD** jumper per Enclosure 5 of AP 1013.

Form ES-401-5 **Written Exam Question Worksheet** Q # 078

Examination Outline Cross-Reference Tier # 1

Evolution/System 040 Steam Line Rupture - Excessive Heat Transfer Group # 1

K/A # 2.1.14 Page # 2-2 RO/SRO Importance Rating 2.5 3.3

Measurement Conduct of Operations: Knowledge of system status criteria which require the notification of plant personnel.
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor trip due to non-bomb steam explosion inside the Intermediate Building.
- Train A and Train B ES actuations due to low RCS pressure.
- All 3 Emergency Feedwater pumps are operating.
- EOP-001 Reactor Trip immediate actions are complete.
- RCS pressure is 1500 psig, lowering.
- Core exit thermocouple temperature 520 degrees F, lowering at 5 degrees per minute.
- OTSG 1A pressure 400 psig, rapidly lowering.
- OTSG 1B pressure 800 psig, slowly lowering.

Based on these conditions identify the ONE selection that describes:
(1) Controlling procedure to be implemented next.
(2) Plant announcement that meets OS-24 procedural requirements.

A. (1) 1203-24, Steam Leak.
(2) Announce the reactor trip, and direct Auxiliary Operators to report to their post trip stations.

B. (1) OP-TM-EOP-003, Excessive Primary-to-Secondary Heat Transfer.
(2) Announce the reactor trip, and direct Auxiliary Operators to report to their post trip stations.

C. (1) 1203-24, Steam Leak.
(2) Announce:
a. Reactor trip with EFW and ES actuation.
b. Secondary Auxiliary Operator should NOT report to EF-V-30 area.

D. (1) OP-TM-EOP-003, Excessive Primary-to-Secondary Heat Transfer.
(2) Announce:
a. Reactor trip with EFW and ES actuation.
b. Secondary Auxiliary Operator should NOT report to EF-V-30 area.

Technical Reference OS-24 Sections 3.6 and 4.4.5, Pages 4, 21 & 37, Rev. 10.
OP-TM-EOP-003, Excessive PSHT, Entry Conditions, Page 1, Rev. 2.
1203-24, Steam Leak, IA 2.0.B.1, Page 2, Rev. 29.

Open Exam Reference None.

Learning Objective V.E.12.02, V.E.15.01

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because the wrong procedure is identified, and the announcement does not meet OS-24 requirements to include plant conditions relevent to Auxiliary Operator emergency response stations.

Distracter is plausible because it does identify a procedure related to a steam leak, and the announcement described would be acceptable if there were no complications like described in the stem.

B INCORRECT answer because the announcement does not meet OS-24 requirements to include plant conditions relevent to Auxiliary Operator emergency response stations.

Distracter is plausible because the correct procedure is identified.

C INCORRECT answer because the wrong procedure is identified.

Distracter is plausible because the announcement meets OS-24 requirements to include plant conditions relevent to Auxiliary Operator emergency response stations.

D CORRECT answer.

Comments Question addresses Conduct of Operations: Knowledge of system status criteria which require the notification of plant personnel, and 10CFR55.43(b)(5) (Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations), linked to a Main Steam Line Rupture event, and 10CFR55.43(b)(5) (Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations).

OS-24 specifies PLANT ANNOUNCEMENT requirements to include as a minimum any plant conditions relevant to Auxiliary Operator emergency response stations.

Based on stem conditions presenting a steam line rupture, and excessive PSHT, system status required to be announced are EFW actuation with the existence of the steam line rupture in the Intermediate Building (AO would be required to report to the EFW system), and HPI actuation (another AO is required to report to the Makeup Pump area). Controlling procedure to be identified for the conditions is EOP-003, Excessive PSHT.

3.6 EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER (XHT):

XHT is undesired heat removal by one or both OTSGs. XHT can be confirmed if ALL of the following conditions exist:

- RCS average temperature below 540°F
- Uncontrolled lowering of RCS temperature
- Tsat for OTSG pressure is less than Tcold on affected OTSG(s)

3.7 FEEDWATER:

A water source to the OTSG(s) from either the Main or Emergency Feedwater Systems.

3.8 LACK OF PRIMARY-TO-SECONDARY HEAT TRANSFER (LOHT):

LOHT is the inability of either OTSG to remove sensible heat from the RCS. LOHT can be confirmed if one of the following sets of conditions exists:

- Core exit temperatures rising above 580°F **and** at least one RC Pump operating
- Core exit temperatures rising **and** NO FEEDWATER available
- Core exit temperatures rising **and** RCS circulation can not be confirmed

3.9 MINIMIZE SCM:

An intentional reduction of the reactor coolant pressure temperature relationship as close as practical to the 25°F subcooling margin or RCP NPSH limit. Actions to minimize SCM are described in Guide 8.

3.10 OTSG AVAILABLE:

A physical condition where the OTSG demonstrates level and pressure control. It means the OTSG is in a condition where primary to secondary heat transfer would be possible. Primary to secondary heat transfer need not be demonstrated to determine this availability.

- Primary to secondary leakage should not be considered a means of OTSG level control.
- A dry OTSG is not available.
- An OTSG isolated IAW EOP-005 isolation criteria is not available.

4.4.3 The conditions for CARRYOVER STEPS are verbalized to the Control Room team to ensure the step is performed when the condition of the step is satisfied.

4.4.4 The following are approved verbal abbreviations:

4.4.4.1 "EOP (number)" or "AOP (number" in place of OP-TM-EOP-xxx or OP-TM-AOP-xxx

4.4.5 PLANT ANNOUNCEMENTS:

A. After IMA are completed a plant announcement should be made. The announcement is to ensure that all auxiliary operators or any other ops personnel in the plant are aware of plant status. The announcement must include at a minimum any plant conditions relevant to Attachment E "Auxiliary Operator Emergency Response Stations".

B. Plant announcements should be made over the plant page and Ops radio systems.

4.4.6 Crew BRIEFINGS

Operations shift management conducts briefings whenever it is appropriate to involve the entire control room team in a discussion. A brief is used to ensure Control Room team members are aware of plant status and direction or to involve the team in event diagnosis.

A. A brief begins by announcing "Attention for a BRIEF".

B. Team members acknowledge by saying "listening".

C. A Crew brief may include, but is not limited to the following:

- Nature of transient and procedures in use
- Expected plant response and mitigation strategy
- Request for specific plant parameters to validate plant status
- Procedure priority.

D. At the end of the brief, the CRS should reinforce team roles and responsibilities and requests if <u>any</u> team members have questions.

E. A brief ends with the statement "End of BRIEF".

4.4.7 Crew UPDATE

A. An UPDATE is short (usually 10 to 15 seconds) information transfers to the entire control room team. A UPDATE is performed as follows:

- A team member announces "Attention for an UPDATE"

ATTACHMENT E
Auxiliary Operator Emergency Response Stations

Revision 06/17/04	**1. COOLDOWN FROM OUTSIDE CR**	**2. LOSS OF STATION POWER**	**3. REACTOR /TURBINE TRIP**	**4. LOSS OF INSTRUMENT AIR**
SECONDARY SAFE SHUTDOWN	(1) If REQD, TRIP TURB AND MFWPS (2) EFW AREA	EF-V-30'S AREA	(1) IF EFW, EF-V-30'S AREA (2) NO EFW • **OPEN AS-V-8** • FW HEATING	(1) CHECK IA-P-1A, 1B and 2B (2) CHECK IA-V-1 (3) SEARCH for leaks INTERM BLDG (4) STBY EF-V-30'S AREA
FIRE BRIGADE (SCUBA)	(FIRE BRIGADE)	SBO DG	(1) CHECK MSSVs (2) WITH EFW **OPEN AS-V-8** (3) POWDEX AND CONTROL LUBE OIL TEMPS	SEARCH FOR LEAKS ON 281 AUX/FHB
PRIMARY SAFE SHUTDOWN	**MU VALVE ALLEY OPEN MU-V-76A/B**	CHECK MU-P'S	STANDBY at MU PUMP AREA	(1) MU-V-20 (2) SEARCH FOR LEAKS ON 305 AUX/FHB
FIRE BRIGADE (ANSUL)	(FIRE BRIGADE)	EG-Y-1A & B	**FIRE AUX BOILERS**	(1) CHECK IA-P-4, SA-P-1A/B & IA-P-2A (2) SEARCH for Leaks in TB

*Any actions performed prior to CRS concurrence are designated with **BOLD TYPE**.
NOTE: If multiple conditions are present, then respond to the highest priority (e.g. LOOP is 2, RX TRIP is 3)

EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER

1.0 ENTRY CONDITIONS - Excessive Primary to Secondary Heat Transfer (PSHT) while shutdown prior to DHR operation.

2.0 IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 2.A **PERFORM** Rule 3, XHT.	
___ 2.B **INITIATE** Guide 9, "RCS Inventory Control".	

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.1 **ENSURE** announcement of reactor trip over the plant page and radio.	
___ 3.2 **VERIFY** at least one OTSG has stable pressure with level present.	___ **GO TO** OP-TM-EOP-009.
___ 3.3 **PERFORM** Guide 12, to limit RCS heatup and pressurization.	
___ 3.4 **ENSURE** RCS temperature reduction has been terminated.	___ **If** PSHT is not excessive **and** temperature reduction is due to HPI/Break Cooling, **then** **GO TO** OP-TM-EOP-006.
___ 3.5 **VERIFY** primary to secondary heat transfer is being established.	___ **GO TO** OP-TM-EOP-004.
___ 3.6 **VERIFY** RCS Tcold > 525°F.	___ **INITIATE** Emergency boration - Rule 5, EB.
___ 3.7 **ENSURE** performance of an alarm review.	

1.0 **SYMPTOMS**

1. Decreasing secondary steam pressure.

2. Electrical load reducing (mismatch between electrical load and Rx Power).

3. Decrease in pressurizer level, R.C. Pressure, and cold leg temperature.

4. For a leak inside the Reactor Building; Indication of increasing Reactor Building pressure and temperature.

5. For a leak outside the Reactor Building; Noise may be heard in Control Room or a report made from personnel outside the Control Room.

2.0 **IMMEDIATE ACTION**

A. Automatic Action

1. If HSPS MFW Isolation actuates (<600 psig) on the affected OTSG (could be both), the following valves auto close.

   a. Startup Feedwater Control Valve FW-V-16A(B)

   b. Main Feedwater Control Valve FW-V-17A(B)

   c. Main Feedwater Block Valve FW-V-5A(B)

   d. Startup Feedwater Block Valve FW-V-92A(B)

2. Possible Reactor trip on low pressure.

B. Immediate Manual Action

1. If the steam leak is upstream of the turbine stop valves (or the leak location is unknown) and either:

   a. HSPS actuates on either SG or

   b. Continued operation presents a hazard to personnel or equipment required for safe shutdown.

   Then manually trip the reactor and go to OP-TM-EOP-001.

2. If the steam leak is downstream of the turbine stop valves and time permits.

   Then reduce power to < 45 percent and trip the turbine IAW 1102-4.

Form ES-401-5 **Written Exam Question Worksheet** Q # 079

Examination Outline Cross-Reference Tier # 1

Evolution/System 057 Loss of Vital AC Electrical Instrument Bus Group # 1

K/A # AA2.12 Page # 4.2-42 RO/SRO Importance Rating 3.5 3.7

Measurement Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: PZR level controller, instrumentation, and heater indications.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** A.

Initial plant conditions:
- Reactor at Hot Shutdown conditions.
- 120V AC bus ATB de-energized in accordance with 1107-2B, 120 Volt AC Vital Electrical System.
- Maintenance personnel implementing Troubleshooting Plan in 'A' Inverter Room and in the Relay Room. At their request:
  - Pressurizer level control is in MANUAL.
  - Pressurizer heater controls in AUTOMATIC.

Event:
- Loss of 120V AC bus ATA .

Current conditions:
- Pressurizer Level H/A controller indications:
  - Hand and Auto lamps NOT lit.
  - Demand and Measured Variable indications at mid-scale.
- Pressurizer heater control indications:
  - SCR heater controller H/A lamps NOT lit.
  - Pressure controller Demand and Measured Variable indications at mid-scale.
  - Console Right Heater Bank ON/OFF lamps are NOT lit.

Based on these conditions identify the ONE selection below that describes:
(1) Status of Pressurizer level and RCS pressure control systems.
(2) Appropriate method of controlling Pressurizer level.

A. (1) Automatic controls NOT operable.
Manual controls NOT operable.
(2) Use MU-V-217 to control Pressurizer level.

B. (1) Automatic controls operable.
Manual controls NOT operable.
(2) Use Automatic control of MU-V-17.

C. (1) Automatic controls NOT operable.
Manual controls operable.
(2) Use Manual control of MU-V-17.

D. (1) Automatic controls operable.
Manual controls operable.
(2) Use Automatic or Manual control of MU-V-17.

Technical Reference 1202-40, Loss of ICS/NNI Hand and Auto Power, Sections 3.C and 3.D, Pages 3 and 4, Rev. 41.

Open Exam Reference None.

Learning Objective V.D.20.03

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Written Exam Question Worksheet

Question NRC Exam History

Question Cognitive Level

☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer.

B INCORRECT answer because control system status is not correct for loss of ICS/NNI Hand and Auto Power, and MU-V-17 cannot be controlled from the control room under these conditions.

C INCORRECT answer because control system status is not correct for loss of ICS/NNI Hand and Auto Power, and MU-V-17 cannot be controlled from the control room under these conditions.

D INCORRECT answer because control system status is not correct for loss of ICS/NNI Hand and Auto Power, and MU-V-17 cannot be controlled from the control room under these conditions.

Comments None.

2.2 Manual Action

> **CAUTION**
>
> Do not select alternate ICS/NNI Power or otherwise attempt to restore power at this point. Upon restoration of HAND power, main and startup feedwater valves will stroke fully open.

> **CAUTION**
>
> If a. or b. below cannot be performed as written (i.e., ATWS or failure of main turbine stop valves to close) go directly to OP-TM-EOP-001 at that point for direction on performance of remedial actions. Refer to this procedure for additional guidance.

On a confirmed loss of ICS Hand and ICS Auto power.

1. **TRIP** the reactor and **VERIFY** power less than 10%.
2. **TRIP** the main turbine and **VERIFY** T/G stop valves closed.
3. **TRIP** both main feedwater pumps.
4. **GO TO** OP-TM-EOP-001 and refer to this procedure for additional guidance.

> **NOTE**
>
> Control room indications listed in Table 1, are available for controlling plant parameters.

3.0 **FOLLOW UP ACTION**

Objective: The objective of this procedure is to stabilize the plant in a hot shutdown condition and to restore ICS/NNI power. If unable to restore power, proceed with a controlled plant cooldown.

_____ A. **VERIFY** EFW Controls OTSG Level at ≥ 25" startup range.

_____ B. **OPEN** MS-V-4A/B with B/U loaders, ("BACK UP CTRL" Bailey Stations) to reseat main steam safety valves and control OTSG pressure.

C. **IF** MU-V-17 cannot be controlled in Hand or Auto, **THEN** (NA if MU-V-17 can be controlled)

_____ a. **USE** MU-V-217 to control pressurizer level.

_____ b. **DISPATCH** an operator to isolate MU-V-17 locally **BY CLOSING MU-V-91B**.

_____ D. **RESTORE** pressurizer heater operation by:

1. **VERIFY** pressurizer level ≥ 80" on RC-LI-777A.

2. **PLACE** pressurizer level LO-LO interlock switch to the bypass position (Key #2) inside ICS/NNI Pwr Monitoring Cabinet (Key #214).

> **CAUTION**
>
> Turn all heaters "OFF" if level on LI-777A drops below 80".

_____ 3. **OPERATE** pressurizer heater bank control switches on console right to control RCS pressure.

_____ E. **MONITOR** RCP "SEAL WTR TEMP AT RAD BRG", (PPC Pt. Nos. A0521 - A0524), and "SEAL 1 INLET TEMP", (PPC Pt. Nos. A0525 - A0528), on the Plant Process Computer.

F. **IF** needed to defeat and restore letdown and bleed paths, **THEN** proceed as follows:

TO RESTORE Letdown: (N/A if letdown is not required)

_____ 1. **LIFT LEAD** 7-3-3-16 for MU-V-1A/B in the ICS/NNI Cabinet and record on Enclosure 5 of AP 1013.

_____ 2. **LIFT LEAD** 5-4-5-4 for MU-V-3 in the ICS/NNI Cabinet and record on Enclosure 5 of AP 1013.

_____ 3. **CLOSE** MU-V-4 if desired to limit letdown flow. N/A if not required.

> **CAUTION**
>
> MU-V-5 is failed in the mid position. If a slow return to letdown is required, then MU-V-5 will need to be placed in MANUAL and CLOSED or isolated by CLOSING MU-V-97A.

4. **IF** a controlled return of Letdown is desired, **THEN:** (N/A if SM/CRS authorizes a rapid return of letdown)

_____ 1. **DISPATCH** an AO to MU-V-5/97A.

_____ 2. **PLACE** MU-V-5 in MANUAL control and CLOSE. (N/A if using MU-V-97A)

_____ 3. **CLOSE** MU-V-97A (N/A if using MU-V-5).

_____ 4. **OPEN** MU-V-1A and MU-V-1B.

_____ 5. **OPEN** MU-V-3.

_____ 6. **THROTTLE OPEN** MU-V-5 for about 10 gpm (N/A if using MU-V-97A).

Form ES-401-5

Written Exam Question Worksheet

Q # 080

Examination Outline Cross-Reference Tier # 1

Evolution/System E02 Vital System Status Verification Group # 1

K/A # EA2.1 Page # 4.3-3 RO/SRO Importance Rating 2.5 4.0

Measurement Ability to operate and / or monitor the following as they apply to the (Vital System Status Verification): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** D.

Initial conditions:
- Reactor operating at 90% power with ICS in automatic.
- Control rod index 290.
- Makeup Tank water addition in progress.
  - Ten (10) gallons of 100 gallons has been added from RC Bleed Tank A.

Event:
- Reactor trip.
- Safety Rods in Groups 1-4 did NOT trip.
- Both Main Feedwater Pumps tripped.

Current conditions:
- Reactor power is 2%, lowering slowly.
- RCS pressure is 2000 psig, lowering at 30 psig per minute.

Identify the ONE selection below that describes required actions for these conditions.

A. Trip the Main Turbine, and initiate 1202-8, CRD Equipment Failure.

B. Ensure EFW is actuated, and initiate 1203-10, Unanticipated Criticality.

C. De-energize 1G and 1L 480V busses, and terminate the RCS dilution in progress

D. Initiate EOP-010 Rule 5, Emergency Boration, and terminate the RCS dilution in progress.

Technical Reference OP-TM-EOP-001, Reactor Trip, Step 3.3, Page 3, Rev. 5.
OP-TM-EOP-010 Rule 5, Page 9. Rev. 3.

Open Exam Reference None.

Learning Objective PCO-01-EOP001

Question Source ☐ New ☐ Bank ☑ Modified Bank

Question #

Parent Question # QR-PCO-01-EOP001-Q02

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the reactor is shutdown as indicated by reactor power < 7% IAW OS-24. These actions are required if the reactor did NOT shutdown on the reactor trip and Main Feedwater is not available.

Distracter is plausible because this action is in the Response Not Obtained column of EOP-001 and with the safety groups stuck out, the examinee may assume that the reactor is not shutdown.

B INCORRECT because the reactor is shutdown as indicated by reactor power < 7% IAW OS-24. These actions are required if the reactor did NOT shutdown on the reactor trip and Main Feedwater is not available.

Distracter is plausible because this action is in the Response Not Obtained column of EOP-001 and with the safety groups stuck out, the examinee may assume that the reactor is not shutdown.

C INCORRECT because the reactor is shutdown as indicated by reactor power < 7% IAW OS-24. These actions are similar to actions required by OP-TM-EOP-020 if unable to trip the reactor prior to evacuating the control room.

Distracter is plausible because this action will remove power from the safety rods and securing dilution is required as part of the Emergency Boration procedure.

D CORRECT. The reactor is shutdown IAW OS-24 with reactor power < 7%, thus the immediate manual actions can be completed. With the IMAs done, group 1 through 4 rods not on the bottom require initiation of emergency Boration IAW Rule 5 and secure any activities which might cause dilution of the RCS.

Comments Modified Bank QR-PCO-01-EOP001-Q02.

3.0 VITAL SYSTEM STATUS VERIFICATION (VSSV)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3.1 **IAAT** a symptom exists, **then** immediately treat the symptom using the following priority: 1. SCM < 25°F **GO TO** OP-TM-EOP-002. 2. XHT **GO TO** OP-TM-EOP-003. 3. LOHT **GO TO** OP-TM-EOP-004. 4. OTSG tube leakage > 1 gpm **GO TO** OP-TM-EOP-005.	
_____ Time	
_____ 3.2 **ANNOUNCE** Reactor Trip over plant page and radio (include plant conditions sufficient for NLO response per OS-24).	
_____ 3.3 **VERIFY** control rod groups 1 through 7 are fully inserted.	_____ **INITIATE** Emergency Boration per RULE 5 – EB.
_____ 3.4 **VERIFY** MAIN FW Flow to A & B OTSG are each < 0.5 mlb/hr.	_____ **ENSURE** FW-V-5A AND FW-V-5B are stroking closed or are closed.
_____ 3.5 **VERIFY** OTSG level > setpoint.	_____ **INITIATE** RULE 4 – FWC.
_____ 3.6 **VERIFY** ICS/NNI HAND **or** AUTO Power are available.	_____1. **TRIP** both MFW pumps. _____2. **ENSURE** EFW actuation and **INITIATE** Guide 15. _____3. **CONTROL** OTSG pressure using the ADV B/U loaders _____4. **INITIATE** 1202-40, "Loss of ICS Hand and Auto Power".

5

EB

Rule 5
Emergency Boration

IAAT one the following conditions exist:
- Emergency boration is directed by procedure
- reactor is shutdown **and** all control rods are **not** fully inserted
- reactor is shutdown **and** Neutron flux is **not** decreasing as expected

then Emergency Borate as follows:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** a MU pump is operating	**INITIATE** 1203-15 "Loss of MU"
2. Perform one of the following: ____ **OPEN** MU-V-14A ____ **OPEN** MU-V-14B ____ **PERFORM** Guide 1 "Emergency Boration Backup Methods"	
3. **VERIFY** Total Injection (MU, SI & HPI) > 50 GPM.	1. **INITIATE** OP-TM-211-950 to restore letdown. 2. **INITIATE** OP-TM-211-441 to increase letdown flow. 3. **If** MU tank level > 92" or MU tank pressure > 34 psig, **then BLEED** IAW OP-TM-211-462.
4. **STOP** any activities which may be diluting RCS boron concentration.	
5. **If** RCS is subcooled and neutron flux indication is rising, **then STABILIZE** RCS temperature.	
6. **REQUEST** sample and analysis for RCS boron concentration.	
7. **When** 1% dk/k SHUTDOWN has been achieved for the expected plant condition (REFER TO Figure 1 of 1103-4, "Soluble Poison Concentration Control", or 1103-15A, "Reactivity Balance") **or** LPI > 1250 GPM, **then** emergency boration **may** be stopped.	

Form ES-401-5 **Written Exam Question Worksheet** Q # 081

Examination Outline Cross-Reference Tier # 1

Evolution/System E05 Steam Line Rupture - Excessive Heat Transfer Group # 1

K/A # 2.1.32 Page # 2-4 RO/SRO Importance Rating 3.4 3.8

Measurement Conduct of Operations: Ability to explain and apply all system limits and precautions.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** C.

Plant conditions:
- Manual reactor trip due to high RB pressure (OTSG 1A steam leak).
- Reactor Trip EOP immediate actions complete.
- EOP-010 Rule 3, Excessive Heat Transfer complete.
- BOTH isolation phases were performed on OTSG 1A.
- OTSG 1B pressure is 800 psig, steady.

Based on these conditions identify the ONE selection below that describes operability status for turbine driven Emergency Feedwater pump EF-P-1.

A. OPERABLE because Auxiliary Steam can be aligned as a redundant steam supply.

B. OPERABLE because minimium degree of (steam supply) redundancy is maintained by two lines from OTSG 1B.

C. INOPERABLE because all steam sources from OTSG 1A are isolated.

D. INOPERABLE because steam flow to the turbine will be less than design basis assumptions.

Technical Reference Technical Specification 3.4.1.1.a and Basis, Page 3-25 (Amendment 242) and 3-26.b (Amendment 242).

Open Exam Reference None.

Learning Objective IV.C.05.27

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because Auxiliary Steam is not considered in the TS basis/accident analysis.

Distracter is plausible because Auxiliary Steam is available under the conditions stated, and EF-P-1 can be operated thrugh AS-V-4

B INCORRECT because these two lines merge to a single line.

Distracter is plausible because each OTSG does provide two lines to supply steam for EF-P-1

C CORRECT. TS and Bases require separate steam flowpaths from each OTSG.

D INCORRECT because one steam source and flowpath meets design basis assumptions.

Distracter is plausible because one steam supply (OTSG 1A) has been isolated.

Comments None.

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY

Applicability

Applies to the operating status of systems and components that function to remove decay heat when one or more fuel bundles are located in the reactor vessel.

Objective

To define the conditions necessary to assure continuous capability of DHR.*

Specification

3.4.1 Reactor Coolant System (RCS) temperature greater than 250 degrees F.

3.4.1.1 Three independent Emergency Feedwater (EFW) Pumps and two redundant flowpaths to each Once Through Steam Generator (OTSG) shall be OPERABLE ** with:

a. Two EFW Pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW Pump capable of being powered from two OPERABLE main steam supply paths.

(1) With one main steam supply path inoperable, restore the inoperable steam supply path to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 12 hours.

(2) With one EFW Pump or any EFW flowpath inoperable, restore the inoperable pump or flowpath to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours.

(3) With one main steam supply path to the turbine-driven EFW Pump and one motor-driven EFW Pump inoperable, restore the steam supply or the motor-driven EFW Pump to OPERABLE status within 24 hours or be in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 12 hours.

(4) With more than one EFW Pump or both flowpaths to either OTSG inoperable, initiate action immediately to restore at least two EFW Pumps and one flowpath to each OTSG:

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* These requirements supplement the requirements of Specifications 3.1.1.1.c, 3.1.1.2, 3.3.1 and 3.8.3.

** HSPS operability is specified in Specification 3.5.1. When HSPS is not required to be OPERABLE, EFW is OPERABLE by manual control of pumps and valves from the Control Room.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal DHR is by the OTSGs with the steam dump to the condenser when RCS temperature is above 250 degrees F and by the DHR System below 250 degrees F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the OTSG is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the OTSGs is provided by the main feedwater system.

The Emergency Feedwater (EFW) System supplies adequate feedwater to the OTSGs at accident pressures, removing heat from the Reactor Coolant System (RCS) to support safe shutdown of the reactor when the normal feedwater supply is unavailable. EFW is not required for normal plant startup and shutdown.

The turbine-driven EFW Pump and two motor-driven EFW Pumps take suction from the Condensate Storage Tanks (CSTs) and deliver flow to a common discharge header. Flowpath redundancy is provided for those portions of the EFW flowpath containing active components between the pumps and each of the OTSGs. Each EFW line to an OTSG includes two redundant flowpaths, each equipped with an automatic control valve (EF-V-30A/B/C/D) and a manual isolation valve (EF-V-52A/B/C/D). Each redundant flowpath is capable of providing adequate flow to the associated OTSG. Heat removed from the OTSGs returns to the Main Condenser through the Turbine Bypass Valves (TBVs) or discharges to the atmosphere through the Main Steam Safety Valves (MSSVs) and/or the Atmospheric Dump Valves (ADVs). An unlimited supply of river water to the EFW Pumps is available using either of the two Reactor Building Emergency Cooling Water (Reactor River Water) Pumps (RR-P-1A/B).

Redundant main steam supply paths are provided to the turbine-driven EFW Pump for certain events involving loss of one steam supply (e.g., main steam and feedwater line breaks). An operable Main Steam supply path delivers steam to the turbine-driven EFW Pump upon HSPS actuation or by operator action from the control room when HSPS is not required. During low pressure conditions, additional steam supply paths from Main Steam (MS-V-10A/B) or Auxiliary Steam can be made available to the turbine-driven EFW Pump as necessary.

During design basis events the EFW System can withstand any single active failure and still perform its function. The limiting design basis accident for the EFW System is a loss of feedwater event with off-site power available. In the event of a loss of all AC power, which assumes multiple single failures, the turbine-driven EFW Pump alone delivers the necessary EFW flow. Consideration of additional failures in the EFW System or Heat Sink Protection System (HSPS) is not required for this event. Additionally, the EFW System capabilities are sufficient to deliver the required flow in licensing basis events (e.g., ATWS failure to trip events, Generic Letter 81-14 seismic events, and the Station Blackout event).

The most limiting EFW flow requirement is met when at least two EFW Pumps are operable and at least one EFW flowpath to each OTSG is operable. When three pumps and two flowpaths to each OTSG are operable, the EFW System can withstand any single active failure. Examples of single active failures include: failure of any one EFW Pump to actuate, failure of one HSPS train to actuate, or failure of one redundant flowpath to either OTSG. Initially after a shutdown, any two EFW Pumps are required to remove RCS heat with one pump eventually sufficing as the decay heat production rate diminishes.

Form ES-401-5 **Written Exam Question Worksheet** Q # 082

Examination Outline Cross-Reference Tier # 1

Evolution/System 028 Pressurizer Level Malfunction Group # 2

K/A # AA2.10 Page # 4.2-23 RO/SRO Importance Rating 3.3 3.4

Measurement Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Whether the automatic mode for PZR level control is functioning properly, necessity to shift to manual modes.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor operating at 100% power with ICS in full automatic.

Event:
- Reactor trip from 100% power.
- RCS letdown flow 70 gpm.

Current plant conditions:
- EOP-001 Reactor Trip immediate actions complete.
- All Pressurizer level channels at 180 inches, rising slowly.
- Makeup Tank level 85 inches, lowering slowly.
- Makeup flow 50 gpm.
- RCP Seal Injection flow 38 gpm.

Based on these conditions identify the ONE selection below that describes required action.

A. Isolate makeup flow using EOP-010 Guide 9, RCS Inventory Control.

B. Verify Pressurizer level stabilizes between 200-240 inches IAW EOP-001, Reactor Trip.

C. Select a Bleed Tank and shift MU-V-8 to the BLEED position using OP-TM-211-462, Lowering RCS/MU Volume - Bleed.

D. Transfer MU-V-17 control to HAND and maintain Makeup Tank level constant using 1202-29, Pressurizer System Malfunction.

Technical Reference OP-TM-EOP-001, Reactor Trip, step 3.8, Page 5, Rev. 5.
EOP-010 Guide 9, RCS Inventory Control, Step B.2.2, Page 24, Rev. 3.

Open Exam Reference

Learning Objective V.E.21.03

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. MU-V-17 is not properly positioned for the conditions due to mechanical problem with the valve or a controller problem. This is step 2 to be performed if Pressurizer level is higher than desired.

B INCORRECT because the RO adjusts the level setpoint to the "desired value" (90-110 inches) after a trip.

Distracter is plausible since this is desired value specified in Guide 9 for other conditions.

C INCORRECT because Guide 9 requires MU-V-217 and MU-V-17 to be closed or isolated.

Distracter is plausible because this action would reduce Pressurizer level.

D INCORRECT because 1202-29 is for instrument failures.

Distracter is plausible because of other 1202-29 guidance for manual pressurizer level control in response to level instrument failures.

Comments None.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 3.7 **VERIFY** 1D **and** 1E 4160V buses are energized from auxiliary transformers.	_____ **If** neither 1D **nor** 1E is energized, **then INITIATE** OP-TM-864-901 to supply an ES Bus _____ **If** offsite power has been lost, **then INITIATE** OP-TM-AOP-020, "Loss of Station Power".
_____ 3.8 **VERIFY** PZR Level **and** MU Tank Level are being controlled.	_____ **INITIATE** Guide 9 RCS Inventory Control.
_____ 3.9 **ENSURE** OTSG pressure is being controlled at desired values using TBVs/ADVs.	_____ **INITIATE** Guide 6 OTSG Pressure Control.
_____ 3.10 **VERIFY** RCS pressure is trending toward desired post trip condition.	_____ **INITIATE** Guide 8, RCS Pressure Control.
_____ 3.11 **VERIFY** <u>both</u> Generator Breakers are OPEN.	**IAAT** generator MW ≤ zero **or** turbine speed < 1770 RPM, **then** _____ **OPEN** GB1-12 _____ **PLACE** "Emergency Rev PWR Bypass" switch in BYPASS **and OPEN** GB1-02.
_____ 3.12 **VERIFY** the Generator field breaker is OPEN.	**IAAT** GB1-12 **and** GB1-02 are OPEN, **then** _____ **OPEN** the Generator Field Breaker
_____ 3.13 **VERIFY** primary **and** secondary Instrument Air pressure > 80 psig.	_____ **INITIATE** 1202-36. "Loss of Instrument Air".

Guide 9
RCS Inventory Control

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A. MU Tank Level Control	
1. **VERIFY** MU Tank Level > 55 inches.	1. **OPEN** MU-V-14A **or** MU-V-14B 2. **When** MU Tank level > 60" **and** letdown flow > makeup flow, **then CLOSE** MU-V-14A and MU-V-14B.
2. **VERIFY** MU Tank Level < 96 inches.	**If** letdown is **not** isolated, **then INITIATE** OP-TM-211-462, "Lowering RCS/MU Volume – Bleed".

NOTE : "Desired" pressurizer level is determined as follows:
If reactor is critical, **then** "desired" pressurizer level is 200 to 240 inches
If reactor is **not** critical **and** SCM > 25°F **and** RCS > 329°F **and** steam bubble in the PZR, **then** "desired" pressurizer level is 90 to 110 inches
In other plant conditions there is no specified "desired" pressurizer level.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B. Pressurizer Level Control	
1. **VERIFY** Pressurizer Level is above "desired" level.	____ 1. **If** a MU pump is **not** operating, **then INITIATE** 1203-15. ____ 2. **ENSURE** MU-V-5 is CLOSED. ____ 3. **ENSURE** MU-V-17 is OPEN. ____ 4. **THROTTLE OPEN** MU-V-217 ____ 5. **If** MU-V-217 must remain OPEN to maintain desired PZR level, **then CLOSE** MU-V-3. ____ 6. **If** Pzr level **cannot** be maintained with MU-V-217 open **and** MU-V-3 closed, **then INITIATE** HPI
2. **VERIFY** Pressurizer Level is below "desired" level	____ 1. **THROTTLE HPI (Rule 2)**. ____ 2. **ENSURE** MU-V-217 **and** MU-V-17 are CLOSED or ISOLATED. ____ 3. **If** letdown is isolated, **then INITIATE** OP-TM-211-950 to restore letdown flow. ____ 4. **RAISE** letdown flow IAW OP-TM-211-441, "Increased letdown flow rates".

Form ES-401-5 **Written Exam Question Worksheet** Q # 083

Examination Outline Cross-Reference Tier # 1

Evolution/System 051 Loss of Condenser Vacuum Group # 2

K/A # 2.1.33 Page # 2-4 RO/SRO Importance Rating 3.4 4.0

Measurement Conduct of Operations: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2/.3

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** A.

Identify the ONE selection below that describes a plant condition that has Technical Specification required action(s).

A. RCS cooldown, Tave 270 degrees F.
Main condenser vacuum 22 inches Hg.

B. Reactor at hot shutdown conditions.
Motor breaker trips when operator manually starts MU-P-1A.

C. Reactor operating at 10% power.
NI-3 Log Amp failure.

D. Reactor operating at 100% with ICS in full automatic.
Pressurizer level 330 inches.

Technical Reference Technical Specification 3.4.1.1b, Page 3-26, Amendment 242.

Open Exam Reference None.

Learning Objective V.F.01.10

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer.

B INCORRECT answer because Makeup Pump operability for ECCS is not applicable to Technical Specifications if the reactor is subcritical.

C INCORRECT answer because this condition does not require TS action.

D INCORRECT answer because this condition does not require TS action.

Comments This question addresses the KA: Conduct of Operations: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

Notes:

1. Specification 3.0.1 and all other actions requiring shutdown or changes in REACTOR OPERATING CONDITIONS are suspended until at least two EFW Pumps and one EFW flowpath to each OTSG are restored to OPERABLE status.

2. While performing surveillance testing, more than one EFW Pump or both flowpaths to a single OTSG may be inoperable for up to 8 hours provided that:

   (a) At least one motor-driven EFW Pump shall remain OPERABLE, and

   (b) With the reactor in STARTUP, HOT STANDBY, or POWER OPERATION, a designated qualified individual who is in communication with the control room shall be continuously stationed in the immediate vicinity of the affected EFW local manual valves. On instruction from the Control Room, the individual shall realign the valves from the test mode to their operational alignment.

b. Four of six Turbine Bypass Valves (TBVs) OPERABLE. With more than two TBVs inoperable, restore operability of at least four TBVs within 72 hours.

c. The Condensate Storage Tanks (CSTs) OPERABLE with a minimum of 150,000 gallons of condensate available in each CST.

   (1) With a CST inoperable, restore the CST to operability within 72 hours or be in HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours.

   (2) With more than one CST inoperable, restore at least one CST to OPERABLE status or be subcritical within 1 hour, in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

3.4.1.2.1 With the Reactor between 250 degrees F and HOT SHUTDOWN, and having been subcritical for at least one (1) hour, two (2) Main Steam Safety Valves (MSSVs) per OTSG shall be OPERABLE. With less than two (2) MSSVs per OTSG OPERABLE, restore at least two (2) MSSVs to OPERABLE status for each OTSG within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

3.4.1.2.2 With the Reactor between HOT SHUTDOWN and 5% power, and having been subcritical for at least one (1) hour, two (2) MSSVs per OTSG shall be OPERABLE provided the overpower trip setpoint in the RPS is set to less than 5% full power. With less than two (2) MSSVs per OTSG OPERABLE, restore at least two (2) MSSVs to OPERABLE status for each OTSG within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

Form ES-401-5 **Written Exam Question Worksheet** Q # 084

Examination Outline Cross-Reference Tier # 1

Evolution/System 059 Accidental Liquid Radioactive Waste Release Group # 2

K/A # AA2.04 Page # 4.2-45 RO/SRO Importance Rating 3.2 3.5

Measurement Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: The valve lineup for a release of radioactive liquid.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- WDL-T-11A (A WECST) radioactive liquid release has just been started using normal procedures and valve alignments.
- Your Process Instruction Data Sheet did NOT prescribe any abnormal tank/valve configuration for the release flow path.

Event:
- You are performing a verification of the valve alignment to ensure no accidental liquid release is occurring.
- You are verifying the following valve positions:
  - WDL-V-257, WDL-P-14A/B discharge to MDCT.
  - WDL-V-124, WDL-P-14A outlet to the MDCT.
  - WDL-V-125, WDL-P-14B outlet to the MDCT.
  - WDL-V-128, WDL-P14A outlet to the A WECST

Identify the ONE selection below that describes the correct valve alignment for this planned release.

A. WDL-V-257 - Full open.
WDL-V-124 - Throttled for flow control.
WDL-V-125 - Closed.
WDL-V-128 - Full open.

B. WDL-V-257 - Throttled for flow control.
WDL-V-124 - Full open.
WDL-V-125 - Closed.
WDL-V-128 - Full open.

C. WDL-V-257 - Full open.
WDL-V-124 - Closed.
WDL-V-125 - Throttled for flow control.
WDL-V-128 - Full open.

D. WDL-V-257 - Full open
WDL-V-124 - Throttled for flow control.
WDL-V-125 - Closed.
WDL-V-128 - Throttled for flow control.

Technical Reference 1104-29, Liquid Waste Disposal System Section 2.1.11.b, Page 11, Rev. 86.
1104-29S, Transfers From the WECSTs, Section 1.2.24, Page 6, Rev. 68.

Open Exam Reference None.

Learning Objective IV.B.09.04

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer.

B INCORRECT answer because WDL-V-257 is not used as a throttle valve, and WDL-V-124 should be (air loader) throttled to control release flow rate from the tank.

C INCORRECT answer because this line-up could lead to release of the wrong (B) tank.

D INCORRECT answer because WDL-V-128 should be wide open (solenoid operated).

Comments None.

j. Transfer of concentrates from the Reactor Coolant Evaporator Feed Tank to a Reclaimed Boric Acid Tank shall not be made whenever the concentrated boric acid solution in the latter is the source of concentrated boric acid solution for emergency injection to the reactor coolant system.

2.1.11 Transfers from the Waste Evaporator Condensate Storage Tanks

a. Six (6) options exist for disposal of evaporator condensate in the Waste Evaporator Condensate Storage Tanks (WDL-T11A and WDL-T11B) as follows:

1. Transfer to the Reclaimed Water Storage Tank (CA-T6) of the Chemical Addition System.
2. Transfer to the Reactor Coolant Bleed Tank's for re-use in the primary system or for storage.
3. Discharge to the effluent of the Mechanical Draft Cooling Towers.
4. Recycle to the Miscellaneous Waste Evaporator WDL-Z1B for further cleanup.
5. Reprocess a WECST through 1 or 2 WECST Demins to the other WECST.
6. Transfer to CC-T-1/2 or PW-T-1/2 for future processing.

NOTE

Options 1 and 2 are not normally used and requires approval of Director, Operations.

b. All transfers from a Waste Evaporators Condensate Storage Tank shall be preceded by sampling, analysis and authorized by Control Room Supervisor/Shift Manager.

Reactor Coolant Bleed Tanks WDL-T1B and WDL-T1C may be used only for emergency storage of evaporator condensate and if approved by Director, Operations.

Transfers from the Evaporator Condensate Storage Tanks shall utilize Evaporator Condensate Pump WDL-P14A when WDL-T11A is the source tank and Evaporator Condensate Pump WDL-P-14B when WDL-T11B is the source tank unless otherwise directed, by the Control Room Supervisor.

2.1.12 Miscellaneous Transfers

a. Water shall not be transferred from Spent Fuel Pool A to an RC Bleed Tank, while spent fuel is stored in the pool, to below elevation 344' -6" without notifying the Rad Con Department.

_____ Close WDL-V-688

_____ Open WDL-V-694

_____ Open WDL-V-408

_____ Open WDL-V-409

_____ Close WDL-V-410

_____ Close WDL-V-411

_____ 1.2.20 Verify the Test/Normal switch on FR-84 is in the normal position.

_____ 1.2.21 Mark FR-84 chart with the release number and date.

_____ 1.2.22 Verify the setpoint value for the high liquid waste discharge flow interlock is set per 1.2.11.

_____ 1.2.23 Ensure the air pressure is bled off of both WDL-V-124 and WDL-V-125 by:

1. Turning the air loader pressure to "0" psig.
2. Position the "ON/ISOLATE" switch to the "ON" position.
3. Select WDL-V-124 for about 15 seconds to relieve the air pressure.
4. Select WDL-V-125 for about 15 seconds to relieve the air pressure.
5. Turn the valve selector switch to "ISOLATE" position.
6. Turn the ON/ISOLATE switch to the "ISOLATE" position.

_____ 1.2.24 Open/verify open WDL-P-14A suction and recirculation valves.

_____ Open WDL-V-118

_____ Open WDL-V-128

_____ 1.2.25 Open WDL-V-257 as follows:

_____ 1. Verify no alarms on RM-L-6 and RM-L-7.

_____ 2. Turn the "NORM/RESET" key lock switch for WDL-V-257 (located at the radwaste panel) to "RESET" and then release.

_____ 3. Place the "AUTO/CLOSE" selector switch for WDL-V-257 (located at the radwaste panel) in the "AUTO" position.

_____ 4. Verify that WDL-V-257 is open by observing the position indicator lights at the radwaste panel.

Examination Outline Cross-Reference Tier # 1

Evolution/System E08 LOCA Cooldown Group # 2

K/A # 2.2.25 Page # 2-7 RO/SRO Importance Rating 2.5 3.7

Measurement Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- RCS LOCA caused ESAS actuation.
- Operating crew implementing EOP-006, LOCA Cooldown.
- HPI Pumps were NOT secured in accordance with TS requirements.
  - 10CFR50.54X was invoked.
- RCS subcooled margin 2 degrees.
- All 4 HPI Valves MU-V-16A-D wide open.

Event:
- Pressurizer level begins to rise rapidly due to RCS system re-filling.
- RCS pressure is rising rapidly, approaching the TS 3.1-1 limit on EOP-010 Figure 1.
- Operator is experiencing difficulty gaining control of the HPI Valves.
- EOP-010 Guide 23, RCS Pressure & Temperature Limits directs you to open RC-RV-2 (PORV).

Based on these conditions, identify the ONE selection below that describes the basis for directing this pressure reduction to prevent exceeding the limits of the referenced curve.

A. Prevent Pressurizer Relief Valves from opening.

B. Prevent non-ductile fracture of the reactor vessel.

C. Prevent Reactor Coolant System pressure from exceeding 110% of design pressure.

D. Limit RCS subcooled margin to 250 degrees F to restrict cyclic stresses on RCS components.

Technical Reference Technical Specification 3.1.2 Pressurization Heatup and Cooldown Limitations, Bases, first paragraph, last sentence, Page 3-5, Amendment 234.

Open Exam Reference None.

Learning Objective V.F.10.10

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because the reason stated is not the basis for this operating restriction.

Distracter is plausible because the reason listed is associated with high RCS pressure conditions and maintaining RCS pressure boundary integrity.

B CORRECT answer.

C INCORRECT answer because the reason stated is not the basis for this operating restriction.

Distracter is plausible because the reason listed is associated with high RCS pressure conditions and maintaining RCS pressure boundary integrity.

D INCORRECT answer because the reason stated is not the basis for this operating restriction.

Distracter is plausible because the reason listed addresses temperature effects on stress.

Comments None.

Based on the predicted RT_{NDT} after 29 effective full power years of operation, the pressure/temperature limits of Figure 3.1-1 and 3.1-2 have been established by FTI calculation, Reference No. 7, in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2 and ASME Code Section XI, Appendix G, as modified by ASME Code Case N-640 and N-588. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figure 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in RCS "A" loop hot leg. RCS "A" is most conservative and bounds use of "B".

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits.

(5) FTI Calculation No. 32-5011059-00, "TMI-1 Reactor Vessel Adjusted RTNDT Values for 23 and 29 EFPY."

(6) FTI Calculation No. 86-5010023-00, "TMI Cycle 5-11 Final Report."

(7) FTI Calculation No. 32-5011638-02, "TMI-1 29 EFPY P/T Limits."

Form ES-401-5 **Written Exam Question Worksheet** Q # 086

Examination Outline Cross-Reference Tier # 2

Evolution/System 004 Chemical and Volume Control System Group # 1

K/A # A2.02 Page # 3.1-18 RO/SRO Importance Rating 3.9 4.2

Measurement Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of PZR level (failure mode).

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** D.

Plant conditions:
- Reactor operating at 100% power, with ICS in full automatic.
- MU-V-17 in automatic.

Event:
- Selected Pressurizer Level Indication FAILS HIGH over a 1-second period.

Based on these conditions, identify the ONE selection below that describes:
(1) Automatic response.
(2) Controlling procedure.

A. (1) MU-V-17 position remains the same, actual Pressurizer level is not affected.
(2) OP-TM-L2278, SASS Actuation.

B. (1) MU-V-17 opens, actual Pressurizer level rises.
(2) 1202-29, Pressurizer System Failure.

C. (1) MU-V-17 opens, actual Pressurizer level rises.
(2) OP-TM-MAP-G0205, Pzr Level Hi/Lo.

D. (1) MU-V-17 closes, actual Pressurizer level lowers.
(2) 1202-29, Pressurizer System Failure.

Technical Reference 1202-29, Pressurizer System Failure, Page 14, Rev. 60.

Open Exam Reference None.

Learning Objective IV.E.09.02

Question Source ☐ New ☐ Bank Question #

☑ Modified Bank Parent Question # QR4A09-06-Q01

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because Pressurizer level indication is not SASSed, therefore the controlling Pressurizer level will change resulting in MU-V-17 repositioning.

Distracter is plausible because if Pressurizer Level was SASSed then failing high over 1 second would result in swapping to the alternate instrument thus having no effect on MU-V-17 position and pressurizer level.

B INCORRECT because this is opposite to the actual reaction of MU-V-17 and pressurizer level to conditions given in the stem.

Distracter is plausible because this is the correct reaction of MU-V-17 and pressurizer level if the instrument were to fail Low rather than High. Also, the procedure used in the second part of the answer is the correct procedure.

C INCORRECT because this is opposite to the actual reaction of MU-V-17 and pressurizer level to conditions given in the stem.

Distracter is plausible because the instrument failure would result in MAP G-2-5, PZR level Hi/Lo and the procedure listed is the Alarm Response Procedure for MAP G-2-5.

D CORRECT because:
1) Pressurizer Level is not SASSed, therefore the failed instrument will cause MU-V-17 to close and have the indicated effect on Pressurizer level.
2) The procedure listed is the correct procedure for a malfunction in Pressurizer level indication.

Comments None.

SECTION D

Malfunction In Pressurizer Level Indication Or Control

1.0 **SYMPTOMS**

1.1 Disagreement between pressurizer level indicators (computer and console) of more than 12 inches. RC1-LT1 (C1720), RC1-LT3 (C1722) and RC-LI-777.

1.2 Rapid change in indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction, of the pressurizer.

1.3 Possible high or low pressurizer level alarms.

- G-1-5, Pzr Level Hi-Hi
- G-2-5, Pzr Level Hi/Lo
- G-3-5, Pzr Level Lo-Lo

1.4 Pressurizer level indicator(s) **NOT** responding to changes in pressurizer level.

1.5 Hi makeup flow alarm (D-3-1, MU Flow Hi).

1.6 Pressurizer temperature fails to agree with saturation temperature for RCS pressure.

1.7 RCS pressure changes does **NOT** agree with PZR level changes.

2.0 **IMMEDIATE ACTION**

2.1 Automatic Action

2.1.1 If indication fails low

a. Pressurizer heaters trip at 80 inches.

b. Makeup valve MU-V-17 opens.

2.1.2 If indication fails high

a. Makeup valve MU-V-17 closes.

2.2 Manual Action

2.2.1 **TAKE** MU-V-17 under hand control AND **ADJUST** makeup flow to equal letdown flow minus seal injection to maintain makeup tank as constant as possible.

2.2.2 **SELECT** alternate pressurizer level transmitter.

2.2.3 **SELECT** alternate pressurizer temperature transmitter.

Form ES-401-5 **Written Exam Question Worksheet** Q # 087

Examination Outline Cross-Reference Tier # 2

Evolution/System 006 ECCS Group # 1

K/A # 2.4.30 Page # 2-14 RO/SRO Importance Rating 2.2 3.6

Measurement Knowledge of which events related to system operations/status should be reported to outside agencies.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor is operating at 100% power with ICS in full automatic.

0900 Event:
- Operator discovered 1M DC Bus power supply selected to B DC System.
  - Discharge header disconnects are open between MU-P-1A and MU-P-1B.
- 1M DC Bus power supply was promptly selected to A DC System.

0930 Event:
- Overflow of Decay Heat Surge Tank DC-T-1B during operation of DR-P-1B to support planned radioactive liquid release to the river.
  - Decay Heat Services Cooler 2A is out of service due to vibration-induced tube failures.
- DC-T-1B water overflow stopped when DR-P-1B was secured.

Based on these conditions identify the ONE selection below that describes:
(1) Event requiring EARLIEST notification to the NRC.
(2) Actions to comply with NRC notification REQUIREMENTS.

A. (1) 0900 Event.
(2) Contact NRC Operations Center via ENS.

B. (1) 0930 Event
(2) Contact NRC Operations Center via ENS.

C. (1) 0900 Event.
(2) Inform the resident inspector by telephone.

D. (1) 0930 Event
(2) Inform the resident inspector by telephone.

Technical Reference LS-AA-1020, Reportability Reference Manual, F-aa, Page 4, Rev. 6.
OP-TM-211-000, Section 4.6, Page 14, Rev. 5.

Open Exam Reference None.

Learning Objective VII.D.03.05

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because the wrong event is identified.

Distracter is plausible because NRC contact via ENS is correct.

B CORRECT answer.

C INCORRECT answer because the wrong event is identified, and the method of notification and the NRC contact described are not correct.

Distracter is plausible because of the resident inspector is normally readily available and he is an NRC representative.

D INCORRECT answer because the method of notification and the NRC contact described are not correct.

Distracter is plausible because the correct event is identified,

Comments None.

REPORTABILITY REFERENCE MANUAL
VOLUME 1 - TABLE SAF

ID #	REPORT/SUBJECT	REQUIREMENTS	RECIPIENT	DATE DUE & METHOD OF REPORTING	CONTENT	EVENT NUMBER
F-09	Any instance of: (A) A defect in any spent fuel storage structure, system, or component which is important to safety. (B) A significant reduction in the effectiveness of any spent fuel storage confinement system during use.	10CFR72.75(c)(1) 10CFR72.75(c)(2)	NRC Operations Center	ENS within 8 hours. Written report required by 10CFR72.75. (See T-29)	(i) Caller's name and call back telephone number. (ii) Description of the event, including date and time. (iii) Exact location of event. (iv) Quantities and chemical and physical forms of the spent fuel or HLW involved (v) Any personnel radiation exposure data.	1.20
F-10	DELETED					
F-aa	Initiation of any nuclear plant shutdown required by the Technical Specifications.	10CFR50.72(b)(2)(i)	NRC Operations Center	ENS within 4 hours. Written report required by 10CFR50.73 if shutdown is completed. (See T-01)	Same as I-01	SAF 1.2
F-bb	Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	10CFR50.72(b)(2)(iv)(A)	NRC Operations Center	ENS within 4 hours. Written report required by 10CFR50.73. (See T-07)	Same as I-01	SAF 1.5
F-cc	Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the	10CFR50.72(b)(2)(iv)(B)	NRC Operations Center	ENS within 4 hours. Written report required by 10CFR50.73. (See T-07)	Same as I-01	SAF 1.6

4.6 Anytime HPI must be operable, the power source to the 1M 125 VDC distribution panel should be selected to the A DC system if the MU pump discharge cross connects are open between the A & B pumps (i.e., MU-V-77A/B are OPEN and MU-V-76A & B are CLOSED). If the discharge cross connects are open between the B & C pumps (MU-V-76A/B are OPEN and MU-V-77 A/B are CLOSED) then the IM DC panel should be powered from the 1B DC panel. (Reference: FSAR, Sec. 6.1.3.1). Record any misalignment in the Control Room log, minimize misalignment time, and ensure that all misalignments are for less than 72 hours. (T.S. 3.3.2)

4.7 HPI Flow Indication (MU-FI-1126, 1127, 1128, & 1129) is required to be operable for the associated HPI Train to be operable, if that HPI train is also lined up to provide seal injection. When a HPI Flow Indicator is not operable, ensure compliance with Tech Spec 3.3.2 and initiate a 72 HR TS time clock. With authorization from the Operations Director, the MU pump discharge cross connect valve lineup may be swapped to align an inoperable HPI Flow Indicator with the HPI Train not lined up to seal injection. An inoperable HPI Flow Indicator associated with an HPI train not lined up to provide seal injection does not affect HPI operability per Tech Spec 3.3.

4.8 Prompt isolation of the letdown line in the event of a pipe break between the containment wall and the block orifice minimizes the release of high energy fluid into the Auxiliary Building. To accomplish this function, MU-TS-1 and MU-TS-2 along with there associated circuitry, will close MU-V-3 and MU-V-2A & B respectively.

- If either of these circuits is not operable, then initiate a 30 day administrative time clock.
- If both of these circuits are not operable, then enter a 72 hour administrative time clock until at least one of the circuits is placed back in operation.
- A station risk analysis is required if operability cannot be achieved within the respective time clock period.

5.0 <u>COMMITMENTS</u>

5.1 **CM-1**, 1980T0068, NUREG 0680: TMI-1 Restart, Shift Foreman Approval Prior SP Testing Of SR Systems

5.2 **CM-2**, 1982T0049, PSC 81-010: Makeup Line Break (MUT Lo press alarm 18 psig normal 15-35psig)

5.3 **CM-3**, 1982T0053, IR 82-17: Inspection Report 82-17 OI 82-17-01, Check Indicator Light Bulbs Not Normally Lit (ISTs various)

Form ES-401-5 **Written Exam Question Worksheet** Q # 088

Examination Outline Cross-Reference Tier # 2

Evolution/System 064 Emergency Diesel Generator Group # 1

K/A # A2.04 Page # 3.6-10 RO/SRO Importance Rating 3.0

Measurement Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Unloading prior to securing an ED/G.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor triped due to RCS LOCA.
- Loss of off-site power (LOOP).
- All ECCS equipment operating.
- Off-site power has been re-established to Bus 1D.
- Emergency Diesel Generator EG-Y-1A loaded to 1.8 MW.

Based on these conditions identify the ONE selection below that describes actions required to shutdown EG-Y-1A and the basis for those actions.

A. Slowly reduce load to approximately 0.3 MW then open the generator breaker to prevent a reverse power trip using OP-TM-861-901, Diesel Generator EG-Y-1A Emergency Operations.

B. Raise load to approximately 3 MW then run the diesel at that load for one hour to reduce unburned fuel oil, lube oil, and carbon in the cylinders using 1107-3, Diesel Generator.

C. Reduce load to ZERO MW and then trip the diesel within 5 minutes to prevent blower failure; then verify alignment for ES Standby using 1303-4.16, Emergency Power System.

D. Trip the diesel to prevent simultaneous alignment of both diesel generators to the grid when AOP-020, Loss of Station Power, Attachment 2 – Restoration of Off-Site Power, has been completed.

Technical Reference OP-TM-EOP-006, LOCA Cooldown, Step 3.9, Page 3, Rev. 3.
OP-TM-861-901, Diesel Generator EG-Y-1A Emergency Operations, Step 5.1.4, Page 6, Rev. 4.

Open Exam Reference None.

Learning Objective IV.G.08.10

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. OP-TM-861-901 is the governing procedure in this situation.

B INCORRECT. This is an action in 1107-3, but the diesel has not been running at low load.

Distracter is plausible because normal surveilance testing requires holding 3 MW load for 60 minutes.

C INCORRECT. This is the basis for action at low loads but load is not reduced to zero.

Distracter is plausible because blower problems generally occur at low loads. 1303-4.16 contains guidance for returning to ES standby.

D INCORRECT. Diesel generator is not purposely tripped from high loads unless it should have automatically tripped.

Distracter is plausible because we never parallel both emergency generators to the grid at the same time during testing.

Comments None.

		ACTION/EXPECTED RESPONSE		RESPONSE NOT OBTAINED
___	3.7	**ENSURE** performance of an alarm review.		
___	3.8	**REQUEST** SM evaluate Emergency Action Levels (EALs).		
___	3.9	**VERIFY** Emergency Diesel Generators are in ES standby **or** loaded on bus.	___	**If** unloaded, **then INITIATE** OP-TM-861-901 and OP-TM-861-902, and place Diesel Generators in ES standby, <u>one at a time</u>.
___	3.10	**INITIATE** Guide 20, "PRIOR to Transfer to RB Sump".		
___	3.11	**INITIATE** Rule 5, "Emergency Boration".		
☐	3.12	**IAAT** BWST level < 15 ft, **or** RB flood level > 54", **then BRIEF, and INITIATE** Guide 21, "Transfer to RB Sump Re-circulation".		
☐	3.13	**IAAT** RCS pressure < 185 psig **and** LPI flow > 1250 GPM/ train, **then GO TO** Step 3.26.		
☐	3.14	**IAAT** primary-to-secondary heat transfer does **not** exist **and** core cooldown rate < 40°F/hr, **then GO TO** Step 3.36.		

5.0 RETURN TO NORMAL

5.1. SHUTDOWN EG-Y-1A and RETURN to ES Standby.

5.1.1. **VERIFY** RELIABLE OFFSITE power is available to 1D 4160V bus. ____

5.1.2. **VERIFY** ESAS is **not** actuated. ____

5.1.3. **VERIFY** 1SB-D2 is CLOSED. ____

NOTE
G1-02 should be OPENED immediately after load is reduced to 0.3 MW to prevent generator breaker TRIP on reverse power.

5.1.4. **IF** G1-02 is CLOSED, **then** gradually **REDUCE** GOVERNOR to ~ 0.3MW. ____

5.1.5. **ENSURE** G1-02 is OPEN. ____

5.1.6. **ENSURE** Auxiliary Transformer 1B LTC is in AUTO. ____

5.1.7. **PRESS** EG-Y-1A "EMERG SHUTDOWN EXCITER BREAKER TRIP" button. ____

5.1.8. **PRESS** EG-Y-1A STOP button. ____

5.1.9. **ENSURE** EG-Y-1A controls are in ES standby position:

A. EXCITER control is in AUTO. ____

B. "Manual Voltage Controller" to 45 percent. ____

C. EG-Y-1A START control in AUTO (STANDBY). ____

D. SPEED DROOP set at 0 percent (DG A : on governor). ____

E. "UNIT/PARALLEL" switch in the UNIT position (DG A : local alarm panel). ____

5.1.10. **VERIFY** the following EG-Y-1A indications:

A. SPEED CONTROL "HIGH" light is lit ____

B. DF-T-2A (day tank) level > ¾ (DG A : on day tank). ____

5.1.11. **RESET** the fuel rack (DG A : near Engine Mounted Instrument Panel). ____

5.1.12. **PRESS** EXCITER RESET (DG A: on Engine Mounted Motor Starter Box). ____

5.1.13. **RESET** EG-Y-1A (DG A : on Engine Mounted Instrument Panel). ____

5.1.14. **VERIFY** LOCAL ALARMS are CLEAR (DG A : Alarm Panel Room). ____

Form ES-401-5 **Written Exam Question Worksheet** Q # 089

Examination Outline Cross-Reference Tier # 2

Evolution/System 076 Service Water Group # 1

K/A # 2.2.22 Page # 2-7 RO/SRO Importance Rating 3.4 4.1

Measurement Equipment Control Knowledge of limiting conditions for operations and safety limits.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** A.

Initial plant conditions:
- Reactor operating at 100% power with ICS in full automatic.
- Nuclear River Service Water Pumps NR-P-1A and NR-P-1B have tripped.
- Standby NR-P-1C is operating.
- NR-P-1C discharge pressure is 17 psig, slowly lowering.

Current conditions:
- Nuclear River Service Water (NR) and Secondary Services (SR) River Water systems have been cross-connected.
- NR-P-1C discharge pressure is 25 psig, and slowly rising.

Based on these conditions identify the ONE selection below that describes required actions.

A. Commence and perform a Technical Specification required shutdown per TS 3.01.

B. Enter OP-TM-AOP-005, monitor IPSH Bay levels, and if less than 271 foot elevation then trip the reactor and enter OP-TM-EOP-001, Reactor Trip.

C. Start a third SR pump and reduce reactor power as necessary to limit SCCW temperatures.

D. Throttle SR discharge valve SR-V-2 until pressure indicator SR-PI-134 is approximately 20 psig and monitor SCCW temperatures.

Technical Reference 1202-38, NSRW Failure, Step 3.7, Page 3, Rev. 40.
TS 3.3.1.4 Cooling Water Systems, Page 3-22, Amendment 227.
TS 3.01 General Action Requiremenst, Page 3;1, Amendment 98.

Open Exam Reference None.

Learning Objective GLO-14-531

Question Source ☐ New ☑ Bank Question # QR-GLO-14-531-Q01

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. 1202-38 Nuclear Services River Water Failure, requires plant shutdown IAW Tech Spec Section 3.3.

B INCORRECT because entry into AOP-005 from 1202-38 is based on a loss of all NR and SR pumps.

Distracter is plausible because an entry condition into AOP-005 is ISPH Bay level less than 277' with a reactor trip required at 271'.

C INCORRECT because 1202-38 requires shutdown after 1 hour rather than 72 hrs.

Distracter is plausible because Tech Spec. 3.3.2 allows 72 hours with 1 train of Nuclear River inoperable for maintenance or testing.

D INCORRECT because the required action IAW 1202-38 is reactor shutdown. Also, SR-V-2 is throttled to >

21psig.

Distracter is plausible because SR-V-2 is throttled to maintain SR-PI-134 > 21psig.

Comments Bank Question QR-GLO-14-531-Q01 (unmodified).

3.0 **FOLLOW-UP ACTION**

Objective:

Re-establish adequate river water flow for NSCCW and ICCW and protect equipment from damage due to inadequate cooling.

_____ 1. IF non-ES selected NR pump cannot be started, THEN verify reset or reset 27/86 lockout relays for 1R & 1T buses on PCR [or locally on 1R & 1T buses] AND attempt to start second NR Pump.

_____ 2. IF only one NR pump is operating, THEN reduce to two NS coolers by closing NR-V-16A/B/C/D as required AND verify that NR-V-4A & B are CLOSED.

_____ 3. IF all NR and SR pumps are inoperable, then GO TO OP-TM-AOP-005.

_____ 4. Ensure NR-V-1A(B)(C) is closed for any non-running NR pumps.

_____ 5. IF NR supply pressure is inadequate (NR-PI-217 < 21 psig) and no additional NR pumps can be started, THEN cross-connect the SR system to the NR system as follows:

a. Start the third SR pump, if available.

b. OPEN NR-V-6 (SR to NR Cross-tie Valve in HX Vault). [1B ESV MCC Unit 10D]

c. IF NR-V-6 cannot be opened, THEN OPEN NR-V-2 and NR-V-7 (NR to SR redundant Cross-tie valve in IPSH).

d. Throttle SR-V-2 until SR-PI-134 (console left) $\geq$ 21 psig.

e. Reduce plant power if needed to maintain SCCW temperatures.

_____ 6. IF NR supply pressure is inadequate (NR-PI-217 < 21 psig) AND a supply piping problem is suspected (i.e., piping failure or blockage between IPSH and HX Vault), THEN cross-connect the SR system to the NR system as follows:

a. Start the third SR pump, if available.

b. CLOSE NR-V-5 (NR Supply Valve in HX Vault). [1A ES Valves Unit 8D]

c. OPEN NR-V-6 (SR to NR Cross-tie Valve in HX Vault). [1B ESV MCC Unit 10D]

d. CLOSE NR-V-3 (NR Supply Valve in IPSH). [1A ES SH MCC Unit 2B]

e. OPEN NR-V-2 and NR-V-7 (NR to SR redundant Cross-tie valve in IPSH).

f. Throttle SR-V-2 until SR-PI-134 (console left) $\geq$ 21 psig.

g. Reduce plant power if needed to maintain SCCW temperatures.

7. If NR and SR are cross-connected, then commence a reactor shutdown in accordance with Tech Spec 3.3 requirements (3.0.1).

b. CFT boron concentration shall not be less than 2,270 ppm boron. Specification 3.3.2.1 applies.

c. The electrically operated discharge valves from the CFT will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

d. DELETED

e. CFT vent valves CF-V-3A and CF-V-3B shall be closed and the breakers to the CFT vent valve motor operators shall be tagged open, except when adjusting core flood tank level and/or pressure. Specification 3.0.1 applies.

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System

The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The sodium hydroxide (NaOH) tank shall be maintained at 8 ft. ±6 inches lower than the BWST level as measured by the BWST/NaOH tank differential pressure indicator. The NaOH tank concentration shall be 10.0 ±.5 weight percent (%). Specification 3.3.2.1 applies.

c. All manual valves in the discharge lines of the NaOH tank shall be locked open. Specification 3.3.2.1 applies.

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.

a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.

b. Two nuclear service river water pumps must be OPERABLE.

c. Two decay heat closed cycle cooling water pumps must be OPERABLE.

d. Two decay heat river water pumps must be OPERABLE.

e. Two reactor building emergency cooling river water pumps must be OPERABLE.

3.3.1.5 Engineered Safeguards Valves and Interlocks Associated with the Systems in Specifications 3.3.1.1, 3.3.1.2, 3.3.1.3, 3.3.1.4 are OPERABLE. Specification 3.0.1 applies.

3. LIMITING CONDITIONS FOR OPERATION

3.0 GENERAL ACTION REQUIREMENTS

3.0.1 When a Limiting Condition for Operation is not met, except as provided in action called for in the specification, within one hour action shall be initiated to place the unit in a condition in which the specification does not apply by placing it, as applicable, in :

1. At least HOT STANDBY within the next 6 hours.
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the time limits of the specification as measured from the time of failure to meet the Limiting Condition for Operation. Applicability of these requirements is stated in the individual specifications.

Specification 3.0.1 is not applicable in COLD SHUTDOWN OR REFUELING SHUTDOWN.

BASES

This specification delineates the action to be taken for circumstances not directly provided for in the action requirements of individual specifications and whose occurrence would violate the intent of the specification.

Form ES-401-5 **Written Exam Question Worksheet** Q # 090

Examination Outline Cross-Reference Tier # 2

Evolution/System 013 ESFAS Group # 1

K/A # 2.4.49 Page # 2-16 RO/SRO Importance Rating 4.0 4.0

Measurement Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .2

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor tripped due to trip of both Main Feedwater Pumps.
- OP-TM-EOP-001 immediate actions are complete.
- Team is implementing EOP-004, Lack of Primary-to-Secondary Heat Transfer.
- RCS pressure is 2200 psig, rising slowly from the lowest value of 1850 psig during the initial trip transient.
- Core exit thermocouple temperature is 613 degrees F, rising slowly.

Event:
- RCS pressure lowers rapidly to 1685 psig and is stabilizing.
- RB pressure begins to rise at 0.5 psig per minute.
- You, the CRS, announce the transition out of EOP-004.

Based on these conditions identify:
(1) MINIMUM REQUIRED operation of Engineered Safeguards Actuation controls.
(2) Basis for those required actions.

A. (1) Manually actuate BOTH Trains of ESAS.
(2) Operation of ONE HPI/LPI Train and ONE Core Flood Tank will limit clad temperature to 2200 degrees F.

B. (1) Manually actuate BOTH Trains of ESAS.
(2) Operation of ONE HPI/LPI Train and TWO Core Flood Tanks is required to limit clad oxidation to less than 1% of the clad.

C. (1) Manually actuate ONE Train of ESAS.
(2) Operation of ONE HPI/LPI Train and ONE Core Flood Tank will limit clad temperature to 2200 degrees F.

D. (1) Manually actuate ONE Train of ESAS.
(2) Operation of BOTH HPI/LPI Trains and TWO Core Flood Tanks is required to limit clad oxidation to less than 1% of the clad.

Technical Reference Technical Specification 3.3 Bases, Page 3-24, Amendment 227.
OP-TM-EOP-010 Rule 1, SCM, Page 4, Rev. 3.
OP-TM-EOP-010 Guide 2, HPI/LPI Initiation, Page 12, Rev. 3.

Open Exam Reference None.

Learning Objective V.F.01.10

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because BOTH Core Flood Tanks are required to prevent core uncovery.

Distracter is plausible because it states BOTH trains of ES are required to be actuated.

B CORRECT answer.

C INCORRECT answer because BOTH Core Flood Tanks are required to prevent core uncovery, and BOTH trains of ES are required to be actuated.

Distracter is plausible because it is permissible to have one Train of ES inoperable for 72 hours when the reactor is critical.

D INCORRECT answer because BOTH trains of ES are required to be actuated.

Distracter is plausible because it is permissible to have one Train of ES inoperable for 72 hours when the reactor is critical, and the distracter acknowledges the requirement to have BOTH Core Flood tanks operable for the ECCS to meet ESAS Final Acceptance Criteria.

Comments None.

Bases (Contd.)

between 8.0 and 11.0 of the solution sprayed within containment after a design basis accident. The minimum pH of 8.0 assures that iodine will remain in solution while the maximum pH of 11.0 minimizes the potential for caustic damage to mechanical systems and components. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

Maintaining MUT pressure and level within the limits of Fig 3.3-1 ensures that MUT gas will not be drawn into the pumps for any design basis accident. Preventing gas entrainment of the pumps is not dependent upon operator actions after the event occurs. The plant operating limits (alarms and procedures) will include margins to account for instrument error.

The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

The iodine removal function of the reactor building spray system requires one spray pump and sodium hydroxide tank contents.

The spray system utilities common suction lines with the decay heat removal system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.2 and 3.3.3 provided requirements in Specification 3.3.4 are met which assure operability of the duplicate components. The specified maintenance times are a maximum. Operability of the specified components shall be based on the satisfactory completion of surveillance and inservice testing and inspection required by Technical Specification 4.2 and 4.5.

The allowable maintenance period of up to 72 hours may be utilized if the operability of equipment redundant to that removed from service is verified based on the results of surveillance and inservice testing and inspection required by Technical Specification 4.2 and 4.5.

In the event that the need for emergency core cooling should occur, operation of one makeup pump, one decay heat removal pump, and both core flood tanks will protect the core. In the event of a reactor coolant system rupture their operation will limit the peak clad temperature to less than 2,200 °F and the metal-water reaction to that representing less than 1 percent of the clad.

Two nuclear service river water pumps and two nuclear service closed cycle cooling pumps are required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant.

REFERENCES

(1) UFSAR, Section 6.1 - "Emergency Core Cooling System"

(2) UFSAR, Section 14.2.2.3 - "Large Break LOCA"

SCM

1

Rule 1

Loss of Subcooling Margin (SCM)

IAAT SCM < 25°F **and** reactor is shutdown, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** it has been more than two minutes since RCP start	**GO TO** Step 3
2. **ENSURE** all RCPs are shutdown.	**If** all RCPs were **not** tripped within one minute, **then MAINTAIN** RCP(s) still operating until one of the following conditions is satisfied: SCM > 25F LPI flow > 1250 gpm/leg Tclad > 1800°F
3. **ENSURE** 1600 # ESAS has been actuated	**PERFORM** Guide 2 "HPI/LPI Initiation"
4. **ENSURE** EFW has actuated.	
5. **VERIFY** all HPI and LPI components are in the ES condition.	**INITIATE** Guide 3 (LPI) or Guide 4 (HPI).
6. **INITIATE** Guide 15 **and FEED** available OTSGs to 75 to 85% Operating Range Level.	

Guide 2
HPI/LPI Initiation

IAAT Manual ESAS does not function when required, then actuate HPI & LPI as follows:

_____ 1. **OPEN** MU-V-14A **and** MU-V-14B.

NOTE
ES Mode of operations requires two Makeup Pumps with physical separation.

_____ 2. **ENSURE** 2 MU Pumps are operating in the ES mode.

_____ 3. **OPEN** MU-V-16A **and** MU-V-16B **and** MU-V-16C **and** MU-V-16D.

_____ 4. **CLOSE** MU-V-18.

_____ 5. **CLOSE** MU-V-36 **and** MU-V-37.

_____ 6. **START** DC-P-1A **and** DC-P-1B.

_____ 7. **START** DR-P-1A **and** DR-P-1B.

_____ 8. **OPEN** DH-V-5A **and** DH-V-5B.

_____ 9. **START** DH-P-1A **and** DH-P-1B.

_____ 10. **OPEN** DH-V-4A **and** DH-V-4B.

Form ES-401-5 **Written Exam Question Worksheet** Q # 091

Examination Outline Cross-Reference Tier # 2

Evolution/System 035 Steam Generator System (S/GS) Group # 2

K/A # A2.02 Page # 3.4-16 RO/SRO Importance Rating 4.2 4.4

Measurement Ability to (a) predict the impacts of the following mal-functions or operations on the GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor trip/turbine trip.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** C.

Initial plant conditions:
- Reactor operating at 100% power with ICS in full automatic.

Event:
- OTSG 1B develops a large steam rupture.
- Reactor tripped due to low RCS pressure.

Current plant conditions:
- In-core thermocouples 532 degrees F and lowering.
- OTSG 1A pressure 885 psig and lowering slowly.
- OTSG 1B pressure 585 psig and lowering rapidly.
- RB pressure 1.5 psig and slowly rising.

Based on these conditions identify the ONE selection below that describes:
(1) Automatic OTSG level responses.
(2) Applicable procedure for OTSG level control.

A. (1) OTSG 1A stabilizes at Low Level Limits;
OTSG 1B continues to lower to EFW actuation setpoint.
(2) OP-TM-EOP-010 Guide 15, EFW Actuation Response.

B. (1) Both OTSGs lower to EFW actuation setpoint.
(2) OP-TM-EOP-010 Guide 15, EFW Actuation Response.

C. (1) OTSG 1A stabilizes at Low Level Limits;
OTSG 1B continues to lower to EFW actuation setpoint.
(2) OP-TM-EOP-010 Rule 4, Feedwater Control.

D. (1) Both OTSGs lower to EFW actuation setpoint.
(2) OP-TM-EOP-010 Rule 4, Feedwater Control.

Technical Reference OP-TM-EOP-010 Rule 4, Feedwater Control, Page 8, Rev. 3.

Open Exam Reference None.

Learning Objective IV.E.10.02, IV.F.02.04

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the procedure to control level will be EOP-010, Rule 4, rather than Guide 15.

Distracter is plausible because the EFW actuation would send the operator to EOP-010, Guide 15.

B INCORRECT because OTSG 1A will not lower to the EFW setpoint.

Distracter is plausible because the EFW actuation would send the operator to EOP-010, Guide 15.

C CORRECT. OTSG 1A will control at 25 inches (Low Level Limits). OTSG 1B will go to the EFW setpoint because the low OTSG pressure will cause MFW to isolate and OTSG to lower to the EFW setpoint. EOP-010 Rule 4, Feedwater Control contains the guidance to control level at 25 inches.

D INCORRECT because OTSG 1A should not reduce below the EFW actuation setpoint.

Distracter is plausible because OTSG 1B level will reduce to less than the EFW actuation setpoint and EOP-010, Rule 4 is the correct controlling document.

Comments Question blends KA (Ability to (a) predict the impacts of the following mal-functions or operations on the Once Through Steam Generator (OTSG); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor trip/turbine trip) with 10CFR55.43(b)(5) (Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations).

It qualifies as SRO only due to requirement to identify appropriate procedure to correct, control, mitigate.

FWC

4

Rule 4
Feedwater Control

A. **IAAT** the reactor is shutdown, **then:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** SCM > 25°F.	**MAINTAIN** OTSG level 75 – 85% OPERATING Range Level.
2. **VERIFY** at least 1 RCP operating.	**MAINTAIN** OTSG level ≥ 50% OPERATING Range Level.
3. **MAINTAIN** OTSG level ≥ 25" STARTUP Range Level.	

B. **IAAT** OTSG Level < minimum, **then MAINTAIN** the following MINIMUM required flow:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **If** SCM < 25°F **and** both OTSGs are available, **then FEED** > 215 gpm/OTSG using EFW.	**FEED** > 1.0 Mlbm/hr using MFW.
2. **If** SCM < 25°F **and** only one OTSG is available, **then FEED** > 430 GPM to the good OTSG using EFW.	**FEED** > 1.0 Mlbm/hr using MFW.
3. **If** all RCPs are OFF **and** incore temperature is rising, **then FEED** OTSG at maximum available EFW flow.	**FEED** > 1.0 Mlbm/hr using MFW.
4. There is no minimum required flow rate.	

Form ES-401-5 **Written Exam Question Worksheet** Q # 092

Examination Outline Cross-Reference Tier # 2

Evolution/System 001 Control Rod Drive System Group # 2

K/A # A2.15 Page # 3.1-9 RO/SRO Importance Rating 3.6 4.2

Measurement Ability to predict the impacts of the following malfunction or operations on the CRDS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Quadrant power tilt.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- Reactor is operating at 60% power, with ICS in automatic.
- All 4 RCPs are operating.
- Group 7 Rod 1 dropped 40 hours ago.
- Quadrant power tilt exceeds the COLR limit by 6% for the past 3 hours.

Based on these conditions identify the ONE selection below that describes:
(1) Required adjustment to the normal rod position limit.
(2) Actions required if that limit is violated.
(3) Procedure that provides guidance for actions identified in (2) above.

A. (1) RAISE rod index limit.
(2) INCREASE RCS boron concentration to withdraw rods.
(3) 1102-4, Power Operation.

B. (1) REDUCE rod index limit.
(2) REDUCE RCS boron concentration to insert rods.
(3) 1102-4, Power Operation.

C. (1) RAISE rod index limit.
(2) REDUCE RCS boron concentration to insert rods.
(3) 1203-7, Hand Calculations for Quadrant power Tilt and Core Power Imbalance.

D. (1) REDUCE rod index limit.
(2) INCREASE RCS boron concentration to withdraw rods.
(3) 1203-7, Hand Calculations for Quadrant power Tilt and Core Power Imbalance.

Technical Reference 1102-4, Power Operations, Sections 2.2.3, 2.3.1,and 2.4.5.l), Pages 5, 6 and 13, Rev. 107.

Open Exam Reference None.

Learning Objective V.B.04.08

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT answer.

B INCORRECT answer because the rod insertion limit is required to be raised. This action could require rods to be withdrawn further (boron INCREASE would be required).

Distracter is plausible because of common confusion on the rod index limit being a "rod withdrawal" limit, and the procedure identified is correct.

C INCORRECT answer because boron would need to be INCREASED to withdraw rods if the rod insertion limit is not met, and the wrong procedure is identified.

Distracter is plausible for the procedure identified because core quadrant power tilt conditions are included in the stem.

D INCORRECT answer because the rod insertion limit is required to be raised, amd the wrong procedure is identified.

Distracter is plausible because of common confusion on the rod index limit being a "rod withdrawal" limit, and the procedure identified is related to quadrant power tilt - core quadrant power tilt conditions are included in the stem.

Comments Question asks the examinee to describe: (1) whether control rod position limits are raised or lowered to mitigate the effects of quadrant power tilt, and (2) how to control/mitigate the effects. The Examinee is also asked to identify the appropriate procedure for these conditions in accordance with the KA and 10CFR55.43(b)(5).

To answer the question, how does the question ask the examinee to predict the impact of operating with QPT on the CRDS? The Examinee must identify the requirement to raise rod insertion limits, and if the new limit is not met, RCS boron concentration will need to be raised in order to withdraw control rods to an acceptable level in accordance with the new limits.

NOTE

This will ensure the average power (over designated 12 hour period) is less than 2568 MWt.

ii) **MAINTAIN** this reduced reactor power for twice (2x) the time that power was above 100%.

2.1.2 Core Thermal Power is not reduced below 2% reactor power in 1102-4, Power Operation. Reactor shutdown is performed IAW 1102-10. Operation at < 5% NI power should only be a transition. Continuous operation at < 5% power must be specifically authorized by the Operations Director or Shift Operation's Superintendent.

2.1.3 Ensure that rate of reactor power increase is less than limit specified on Enclosure 1 (Mechanical Maneuvering Recommendations).

2.1.4 Regular and frequent cross checking of indicated N.I. power with other thermohydraulic indicators of real power (i.e., core ΔT, MWe, FW flows) can provide early warning of non-conservative NI calibration and/or other process problems requiring resolution.

2.1.5 FW flow correction factors require reset to 1.000 when <u>any</u> of the following conditions for the "power distribution limit" main annunciator window G-2-6 are present:

- Computer Point L3056 in alarm is determined to be a valid indication of excess power
- Power reduction below 85% reactor power
- Less than 4 RC Pumps operating
- A large feedwater chemistry transient
- Δ-T_{cold} setpoint beyond ± 2°F
- T_{ave} setpoint other than 579 ± 0.5°F
- At the direction of the Shift Manager

2.1.6 Do not exceed 75 percent power unless 4 reactor coolant pumps are in operation.

2.2 Core Tilt & Imbalance Limits

2.2.1 During power operation > 50% with 4 reactor coolant pumps running maintain ΔTc less than 5°F.

2.2.2 Axial Power Shaping Rods should be maintained at 30-32% WD except for movement IAW Section 3.4 or when withdrawn at EOC IAW Section 4.1.

2.2.3 Ensure compliance with Quadrant Tilt limits per Tech Spec 3.5.2.4.

2.2.4 Ensure compliance with Axial Power Imbalance limits per Tech Spec 3.5.2.7.2.3.

2.3 Control Rod Position Limits

2.3.1 Maintain control rod positions above the limits specified in the COLR (Tech Spec 3.5.2.5.b).

2.3.2 Ensure control rod group overlap does not exceed 25 +/- 5% (Tech Spec 3.5.2.5.a).

2.4 Other Limits and Precautions

2.4.1 Pressurizer level is maintained within the limits of OP-TM-211-472, Pressurizer Level Control.

2.4.2 RCS Cooldown rate above 525°F is limited to ≤ 100°F/hr.

2.4.3 Ensure Turbine loading, unloading and generator output limits are satisfied IAW OP-TM-301-000, Turbine Generator.

2.4.4 Maintain Generator Reactive Load IAW OP-TM-301-472

2.4.5 Limits on Use of Figures 1, 2, 3, 4 & 6

(a) The volume of concentrated boric acid (2500 ppm or 15000 ppm) can be used from the figure without adjustment if the source concentration is close to the nominal source concentration (2500 ppm or 15000 ppm). "Close" is defined as:

$$\left| \frac{\text{Actual Source Concentration} - \text{Nominal Source Concentration}}{\text{RCS Concentration - Nominal Source Concentration}} \right| < 0.1$$

For greater differences in source boron concentration, inverse proportionally adjust the volume of the addition based on the ratio of the source boron concentration to the nominal source concentration (15000 ppm)

(b) Estimates (i.e. rough interpolation) may be used for initial rod positions between those specified for the curves on Figures 1 & 2.

e) **PERFORM** the actions per Enclosure 2B.

f) **IAAT** IMBALANCE is approaching the "restricted region" of COLR Figures 4, 5 or 6, **then INITIATE** Section 3.4.

g). **IAAT** both OTSGs are on "Low Level Limits", **and** OTSG levels ≤ 25", **then** perform the following:

1. **PLACE** SG/REACTOR DEMAND in HAND.
2. **PLACE** FW ΔTc control in Hand
3. **PLACE** both SG FW A & B DEMAND in HAND.
4. **PLACE** REACTOR DEMAND station in HAND.
5. **LOWER** SG/REACTOR DEMAND to "zero".
6. **LOWER** both SG FW A & B DEMAND to "zero"
7. **ADJUST** REACTOR DEMAND to control reactor power.
8. **LOG:** ICS (621) system is in the "STANDBY" mode.

h). **IAAT** NI power is < 10%, **then**

1. **PLACE** Diamond station in MANUAL.
2. **INSERT** or **WITHDRAW** control rods in sequence to control reactor power.

i) **IAAT** Control rod index is approaching the "restricted region" of COLR Figure 1 or 2, **then ADD** boric acid for a 5% rod withdraw (Figure 3 or Figure 6) IAW 1104-29E.

j) **IAAT** one of the following sets of conditions are satisfied

____ OTSG tube leak > 1 GPM **and** Tavg < 555°F

____ Reactor to be placed in Cold Shutdown and reactor power < 15%.

____ Reactor to be placed in Hot Shutdown and reactor power < 5%.

then

1. **"N/A"** the remainder of this section
2. **INITIATE** Enclosure 2C

k) **MAINTAIN** Generator Reactive Load IAW OP-TM-301-472

Form ES-401-5 **Written Exam Question Worksheet** Q # 093

Examination Outline Cross-Reference Tier # 2

Evolution/System 056 Condensate System Group # 2

K/A # 2.4.6 Page # 2-11 RO/SRO Importance Rating 3.1 3.8

Measurement Knowledge of symptom based EOP strategies.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** C.

Plant conditions:
- Reactor tripped 4 hours ago.
- Due to equipment failures at time of the trip:
  - Only EF-P-2A is operating to supply FW to the OTSGs.
  - FW-P-1A and FW-P-1B are both tripped.
- RCS cooldown in progress.
  - RCS T-Hot is 316 degrees F.
  - RCS pressure is 150 psig.
  - OTSG pressures are 70 psig.
  - OTSG levels have been raised to 40% on the Operating Range.
- CO-P-1A and CO-P-2A are operating.
- Decay Heat Removal system is NOT operating.

Event:
- EF-P-2A trip.
- RCS cooldown has been suspended to conserve OTSG secondary inventory.
  - Turbine Bypass valves have been throttled to maintain RCS pressure and temperature steady.

Based on these conditions identify the ONE selection below that describes:
(1) Method of core cooling to be established.
(2) Applicable procedure.

A. Actuate HPI.
OP-TM-EOP-004, Lack of Primary-to-Secondary Heat Transfer.

B. Initiate HPI COOLING.
OP-TM-EOP-009, HPI Cooling.

C. Feed the OTSGs with Main FW.
EOP-010 Guide 16, EFW Failure.

D. Initiate Decay Heat Removal Cooling.
OP-TM-212-111(112) Shifting DHR Train A(B) from Standby Mode to DHR Operating Mode.

Technical Reference OP-TM-EOP-010 Guide 16, EFW Failure, Page 32, Rev. 3.

Open Exam Reference None.

Learning Objective V.E.21.03

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because the method of core cooling identified is not correct for stem conditions (PSHT still exists), and the procedure identified is also not correct.

Distracter is plausible because the only operating EFP has tripped - and this action is included in EOP-004 prior to establishing HPI COOLING in EOP-009.

B INCORRECT answer because the method of core cooling identified is not correct for stem conditions (PSHT still exists), and the procedure identified is also not correct.

Distracter is plausible because HPI cooling is the method of heat transfer to be used if FW cannot be restored AFTER PSHT no longer exists.

C CORRECT answer. Conditions are met to feed the OTSGs using a Condensate Booster Pump (bypassing the FWPs).

D INCORRECT answer because the method of core cooling identified is not correct for stem conditions (PSHT still exists), and the procedure identified is also not correct. In addition, because of RCS conditions, Decay Heat Removal cooling cannot be initiated.

Comments None.

Guide 16
EFW Failure

IAAT EFW is required **and** all components are **not** functional, **then:**

A. **If** EF-P-1 fails, **then INITIATE** Guide 16.1.

B. **If** EF-P-2A **or** EF-P-2B fails, **then INITIATE** Guide 16.2.

C. **If** EF-V-30A/D **or** EF-V-30 B/C fails, **then INITIATE** Guide 16.3.

D. **If** all of the following conditions exist:

____ EFW is NOT available.

____ FW-P-1A and FW-P-1B are unavailable

____ Condensate Booster Pump is operating,

____ T_c < 488°F (P_{sat} < 600 psig),

then OPEN FW-V-6 to bypass the MFW pumps **and FEED** the OTSGs with MFW.

E. **If** FW is available to OTSG without pressure boundary integrity, **then FEED** that OTSG IAW Guide 13.

Form ES-401-5 **Written Exam Question Worksheet** Q # 094

Examination Outline Cross-Reference Tier # 3

Evolution/System Conduct of Operations Group #

K/A # 2.1.6 Page # 2-1 RO/SRO Importance Rating 2.1 4.3

Measurement Ability to supervise and assume a management role during plant transients and upset conditions.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor tripped due to RCS LOCA.
- OP-TM-EOP-001 immediate actions are complete.
- You are the Control Room Supervisor (CRS), directing implementation of the controlling Emergency Operating Procedure (EOP).
- EOP guidance directs you to INITIATE an additional Emergency Procedure (EP).
  - You hand that EP to a Control Room Operator and direct him to perform this procedure in parallel as you continue with the controlling EOP.

Event:
- The CRO states that Step 14 (check-off space is assigned) of the EP is NOT applicable.

Based on these conditions, identify the ONE selection below that describes:
(1) Actions required prior to proceeding to the next step of the EP.
(2) Documentation requirements this EP step is deemed to be NOT APPLICABLE.

A. (1) Direct the CRO to obtain concurrence from the other CRO prior to proceeding.
   (2) Mark the step NA in accordance with OS-24, Conduct of Operations During Abnormal and Emergency Events.

B. (1) Inform the CRO that after obtaining your concurrence he is required to verbalize the step to the Control Room team to verify non-applicability.
   (2) Mark the step NA in accordance with OS-24, Conduct of Operations During Abnormal and Emergency Events.

C. (1) Direct the CRO to obtain concurrence from the other CRO prior to proceeding.
   (2) Invoke 10 CFR 50.54X.

D. (1) Inform the CRO that after obtaining your concurrence he is required to verbalize the step to the Control Room team to verify non-applicability.
   (2) Invoke 10 CFR 50.54X.

Technical Reference OS-24, Conduct of Operations During Abnormal and Emergency Events, Sections 4.1.9 and 4.1.14.A, Pages 11 and 14, Rev. 10.

Open Exam Reference None.

Learning Objective V.E.12.02

Question Source ☑ New ☐ Bank Question #
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because CROs are not authorized to make this determination.

Distracter is plausible because the documentation method meets procedure requirements for use of NA.

B CORRECT answer.

C INCORRECT answer because CROs are not authorized to make this determination, and 10CFR50.54X should not be invoked for this action.

Distracter is plausible because the procedure involved is not the controlling procedure, and there is a common misconception that emergency procedure problems during accident conditions require NRC notification, invocation of 10CFR50.54X.

D INCORRECT answer because 10CFR50.54X should not be invoked for this action.

Distracter is plausible because the documentation method meets procedure requirements for use of NA, and because of the common misconception that emergency procedure problems during accident conditions require NRC notification, invocation of 10CFR50.54X.

Comments None.

4.1.8 Performing Steps In-Sequence

A. Numbered actions or details contained within a step are designed to be performed in the sequence specified.

B. Lettered or bulleted actions, or details are designed to be performed in any order, but are completed before proceeding to the next numbered step.

C. EVENT PROCEDURE steps are performed in the sequence specified. If the Control Room Supervisor has initiated action from a step, and performance of the next step is not contingent on completion of the original step, then proceeding to the next step is permitted.

D. In the case where procedural limitations are evident during certain conditions that justify performance of steps out-of-sequence from that specified in an EVENT PROCEDURE. Performing steps in a sequence other than that specified in the EVENT PROCEDURE is allowed under the following conditions:

- The CRS reviews all intermediate steps and confirms that the action will not adversely impact the outcome of the EVENT PROCEDURE.

4.1.9 Steps That Do Not Apply

Steps the Control Room Supervisor believes do not apply must be verbalized to the Control Room team to verify applicability prior to proceeding.

4.1.10 Steps previously performed

If directed to perform a procedure which was previously performed (partially or completely), the Control Room Supervisor determines if step must be performed again, and verbalizes this determination to the control room team.

4.1.14 Place keeping in an EVENT PROCEDURE

A. Check-off spaces are checked or otherwise marked after the action required by the step is completed. If the procedure is re-performed, additional marks are used.

B. Check-off spaces for VERIFY steps when used in two column format, are completed as follows. If the condition is satisfied, mark the space for the VERIFY step and leave the right hand column spaces blank. If the condition is not satisfied, leave the VERIFY space blank, and mark the spaces in the right hand column after the action required by the step is complete.

C. 24 Hour clock time should be entered in the TIME spaces which occur periodically throughout the EOP. These reference times are used to perform time dependent actions or to reconstruct the event.

D. EOP Rules posted on the Control Boards contain check-off spaces that are not required to be checked or otherwise marked as the step is performed by Reactor Operators. The check-off spaces are marked afterward as a verification that the Rule was performed correctly.

E. CARRYOVER STEPS are left blank until the step applies, and marked NA after the procedure is completed if the step condition was not satisfied.

4.1.15 TWO COLUMN Format

A. The user of the procedure reads the "ACTION/EXPECTED RESPONSE" from the left hand column.

B. If the action is completed satisfactorily or if the response is as expected, then the user proceeds down to the next step in the left hand column (and skips the right hand "Response not obtained" column)

C. If the <u>action</u> cannot be completed or the response is not as expected, then the user proceeds to the right hand column. The user takes the action described in the right hand column and proceeds to the next step in the left hand column.

D. If a "**VERIFY**" step is used in the LH column and no RNO is specified, then do not proceed past this step if the condition is not satisfied.

Form ES-401-5 **Written Exam Question Worksheet** Q # 095

Examination Outline Cross-Reference Tier # 3

Evolution/System Conduct of Operations Group #

K/A # 2.1.14 Page # 2-2 RO/SRO Importance Rating 2.5 3.3

Measurement Knowledge of system status criteria which require the notification of plant personnel.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** A.

Plant conditions:
- Waste Evaporator Condensate Storage Tank WDL-T-11A, radioactive liquid release in progress.

Event:
- During the release, a CRO notices the following conditions:
  - RM-L-6 HIGH alarm lamp is lit.
  - RM-L-6 Interlock Bypass Switch is in the DEFEAT position.
- RM-L-6 Interlock Bypass Switch is turned to the NORMAL position.
- WDL-V-257 is verified closed.

Based on these conditions, identify the ONE selection below that describes:
(1) Applicable procedure.
(2) Required actions.

A. (1) MAP C-1-1, Radiation Level Hi.
(2) Notify the Group Rad Con Supervisor (GRCS) of the occurrence.

B. (1) MAP C-1-1, Radiation Level Hi.
(2) Direct Chemistry to sample and analyze discharge pit near RM-L-7.

C. (1) 1104-29S, Transfers From the Waste Evaporator Condensate Storage Tanks.
(2) Replace RM-L-6 detector liner and then re-commence the release.

D. (1) 1104-29S, Transfers From the Waste Evaporator Condensate Storage Tanks.
(2) Terminate the release, recording the reason on the release permit, and forward to Rad. Engineering.

Technical Reference MAP C-1-1, Radiation Level Hi, Pages 48 and 49, Rev. 33.

Open Exam Reference None.

Learning Objective IV.B.09.04

Question Source ☐ New ☑ Bank Question # 2003 Audit Q-060
☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT because:
1) Alarm response overrides procedural action IAW OS-24. This is the correct alarm response for the current conditions.
2) The first action per MAP C-1-1 is to verify WDL-V-257 is closed (given in stem) and notify the GRCS.

B INCORRECT because the second part of the answer directs chemistry to sample from the discharge pit when the required sample points are upstream of the WESCT transfer pumps.

Distracter is plausible because the reference given is the correct reference for this scenario and sampling the discharge pit is a logical sample point for effluent discharge.

C INCORRECT because the Alarm Response actions takes priority over the WESCT release procedure.

Distracter is plausible because the procedure used in the first part is the correct procedure for a WESCT release.

D INCORRECT because the Alarm Response actions takes priority over the WESCT release procedure.

Distracter is plausible because the procedure used in the first part is the correct procedure for a WESCT release and the action in the second part is the required action if the release cannot be restarted.

Comments None.

C-1-1
Revision 33

ALARM:

RM-L-6 (RAD. WASTE DISCHARGE)

SET POINTS:

Refer to Operating Procedure 1101-2.1, RMS setpoints

CAUSES:

Hi concentrated liquid waste discharge.

AUTOMATIC ACTION:

Liquid Waste discharge valve to the effluent of the Mech. Draft Cooling Tower (WDL-V-257) closes on Hi Alarm.

OBSERVATION (CONTROL ROOM):

1. RM-L-6 "Alert" alarm on PRF
2. RM-L-6 "Hi Alarm" on PRF
3. RM-L-6 Indication on PRF > above setpoints

MANUAL ACTION REQUIRED:

Alert

1. Evaluate RM-L-6 trend for any obvious and unexpected trend toward Hi Alarm setpoint. Notify GRCS of unexpected trend.
2. If Hi Alarm setpoint is unavoidable, close WDL-V-257 and refer to the Hi Alarm actions below.

NOTE

It is undesirable to continue a release with the alert alarm in constantly.

MANUAL ACTION REQUIRED: (Cont'd)

Hi Alarm:

1. Verify WDL-V-257 closes and liquid release is terminated and notify GRCS of alarm.
2. Flush RM-L-6 per 1104-29S (WECST Release Procedure).
3. Restart the liquid release.
4. If liquid release trips again verify WDL-V-257 closes and take sample at WDL-V-200 if dumping "A" Waste Evap. Cond. Storage Tk. or at WDL-V-201 if dumping "B" Tk. Have Chemistry perform analysis to verify alarm.
5. Check the calculations and confirm the analysis on the Liquid Release Permit. Resample and reanalyze the appropriate tank prior to releasing additional liquid.

Form ES-401-5 **Written Exam Question Worksheet** Q # 096

Examination Outline Cross-Reference Tier # 3

Evolution/System Equipment Control Group #

K/A # 2.2.21 Page # 2-7 RO/SRO Importance Rating 2.3 3.5

Measurement Knowledge of pre- and post-maintenance operability requirements.

(2) Facility operating limitations in the technical specifications and their bases.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** B.

Plant conditions:
- Reactor is at HOT SHUTDOWN.
- Preparing to start up 2 days from now.
- The seals of a pump required to be operable for power operations was replaced today.
- The pump section of the system Technical Specification Surveillance is required for post maintenance testing requirements.
- The normal Tech Spec Surveillance is scheduled for 10 days from today.

Identify the ONE selection below that describes MINIMUM testing requirements to satisfy 1001J.1, Technical Specification Testing Program.

A. ENTIRE surveillance test is REQUIRED to be performed PRIOR TO criticality.

B. The PUMP SECTION of the surveillance test is REQUIRED to be completed prior to criticality.

C. It is permissible to complete the ENTIRE surveillance test following criticality on its normally scheduled date.

D. It is permissible to complete the PUMP SECTION of the surveillance test following criticality on its normally scheduled date.

Technical Reference 1001J.1, Surveillance Test Program, Section 4.5.1.D and 4.5.3, Pages 7 and 8, Rev. 7.

Open Exam Reference None.

Learning Objective VII.B.01.15, V.A.04.04

Question Source ☐ New ☑ Bank Question # NRC 20 Q092

☐ Modified Bank Parent Question #

Question NRC Exam History TMI 2001 Q-092

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because the entire surveillance is not required to be performed prior to criticality. The question specifically asked for the minimum testing requirements.

Distracter is plausible because performance of the entire surveillance would satisfy the requirements of 1001.J.1.

B CORRECT. The pump section must be verified operable prior to the reactor being made critical.

C INCORRECT because the pump section must be verified operable prior to the reactor being made critical.

Distracter is plausible because if the entire surveillance is performed early, then the maintenance/surveillance cycle schedule for the next operating cycle will negatively affected.

D INCORRECT because the pump surveillance must be performed to verify operability prior to criticality.

Distracter is plausible because if the pump surveillance is performed early, then the maintenance/surveillance cycle schedule for the next operating cycle will negatively affected.

Comments 2001 TMI NRC Exam Q-092.

NOTES

1. Maintenance Rule Functional Failure (MPFF) determination will be performed by the System Engineer as part of the IR screening process.

2. Operability and Reportability determinations will be made by Shift Management at the time of the event and validated by the Station Ownership Committee during review of new Issue Reports.

The following table describes the methods to resolve problems encountered during testing:

Problem Description		Method for Resolution
A. Equipment problems	1. Test acceptance criteria cannot be met (i.e. hardware problem prevents performance of test as written or causes test acceptance criteria to be exceeded). This includes alert range Inservice Testing Program results. **OR** Hardware problem is correctable and procedure can be completed as written after repair or adjustment.	**1A. Generate an SDR** **1B. Generate an Equipment Deficiency Tag.** **1C. Generate an Issue Report (IR) to document problem details and to drive an operability/reportability review. Immediate notification of Shift Management is required for prompt review.**
	2. Test acceptance criteria is not effected but a hardware problem prevents test from being completed as written (test may be completed satisfactorily with revised procedure).	**2A. Procedure Change per AD-TM-101-1003.** **2B. Generate an Equipment Deficiency Tag.** **2C. Generate an IR to identify the hardware problem if not previously identified.**
	3. Test acceptance criteria is not effected but hardware problem needs to be corrected. If repair is priority it may be completed during test performance.	**3A. Generate an Equipment Deficiency Tag.** **3B. Generate an IR to document problem details.**
B. Procedure problems	1. Minor issues with procedure quality	**1. HU-TM-104-101-1001**
	2. Substantive Procedure quality issue	**2. Procedure Change per AD-TM-101-1003.**
C. Partial performance (for PMT or "special tests" per section 4.1.4.4)		**Test in accordance with specific provisions given in test** **OR** **Partial test per HU-TM-104-101-1001**
D. Retest after equipment adjustment/repair		**Test in accordance with specific provisions given in test** **OR** **Partial test per HU-TM-104-101-1001**

4.5.2 Surveillance deficiencies shall be identified as they occur, immediately upon discovery of the condition. If testing in a remote location, such as the Weather Station, ISPH, Reactor Building, etc., it is sufficient to immediately notify the Shift Manager of the condition with SDR documentation to follow at a more convenient time.

4.5.3 Retest after the equipment adjustment or repair is completed per 4.5.1.D. If the existing data sheets do NOT lend themselves to documented retesting, a duplicate data sheet, marked as such, shall be submitted with the completed data package;

4.5.4 In every case, necessary equipment repairs must be made in a timely manner, consistent with plant operational requirements. If possible the repairs should be completed before the late performance date even if operational requirements are less restrictive.

4.5.5 If the repairs CANNOT be completed (or if the repair efforts are stopped for any other reason) before the Tech Spec Date, the Issue Report shall be annotated as below:

- The Issue Report should reference the Surveillance Test Number and the Work Order number, and test date.
- The Issue Report priority will be consistent with the identified failure: Deficiencies which render equipment inoperable when required to be operable by Tech Specs will normally be assigned a high priority, while failures of a lesser nature may receive a lower priority. Priority will be determined by the Station Ownership Committee.
- If the repairs CANNOT be completed before the late performance date the surveillance test package will be submitted to the STC who will record the test as being performed on time with outstanding deficiencies. The deficiencies will be recorded in the Closing Remarks of the test record.

4.5.6 If deletion of part of a test is necessary because of plant or equipment conditions, the deleted section must be performed before the system is needed for plant operation. Rescheduling will normally be accomplished manually by the Work Group Supervisor (Step 5.1.4) Equipment Status Tags (EST) as described in 4.6 below shall also be used to insure testing.

4.5.6.1 Clearly document any testing no completed on the Surveillance Test Cover Sheet and in the Work Order CREM.

4.5.7 If equipment tagged with a EDT is NOT repaired before additional testing is required, it is NOT necessary to generate additional SDRs for the same problem. If such a situation exists:

1. Enter the EDT number adjacent to the test step(s) affected by the equipment problem.
2. Document the EDT and A/R numbers in the Additional Test Completion Details of the test package and in the Work Order Completion Remarks.

Form ES-401-5 **Written Exam Question Worksheet** Q # 097

Examination Outline Cross-Reference Tier # 3

Evolution/System Equipment Control Group #

K/A # 2.2.5 Page # 2-5 RO/SRO Importance Rating 1.6 2.7

Measurement Knowledge of the process for making changes in the facility as described in the safety analysis report.

10CFR55.43(b)(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .3

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** C.

Identify the ONE temporary change below that requires processing and approval using CC-AA-112, Temporary Configuration Changes.

A. Installation of rigging to support maintenance.

B. Installation of temporary lead shielding to reduce radiation dose.

C. Installation of an inflatable plug to seal a concrete pipe penetration.

D. Jumper installation to support performance of a surveillance procedure.

Technical Reference CC-AA-112, Temporary Configuration Changes, Attachment 2, Pages 24 and 25, Rev. 8.
Temporary Change Tracking Log item 04-00845.

Open Exam Reference None.

Learning Objective None.

Question Source ☑ New ☐ Bank Question #

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☑ Memory/Fundamental Knowledge ☐ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT answer because installation of rigging to support maintenance is typically addressed by pre-engineered procedures (CC-AA-112 Page 24).

Distracter is plausible because it represents a temporary change to the plant.

B INCORRECT answer because installation plant barriers, including temporary lead shielding for reduction of radiation dose rates, is typically addressed by pre-engineered procedures (CC-AA-112 Page 24).

Distracter is plausible because it represents a temporary change to the plant.

C CORRECT answer.

D INCORRECT answer because jumper installation to support performance of a surveillance procedure is a repetitive action, typically controlled by the surveillance pocedure itself (CC-AA-112 Page 25).

Distracter is plausible because it represents a temporary change to the plant.

Comments None.

ATTACHMENT 2
TCCPs, Exclusions and Associated Administrative Controls (CM 6.1.2.1& CM-6.1.5.3)
Page 1 of 3

Temporary configuration changes are controlled either through TCCPs or through use of procedures that have been pre-engineered. Pre-engineered procedures allow the Installer to place the detailed instructions for implementation, removal and configuration restoration directly into the work package used for performing the work without the need for a TCCP. Pre-engineered procedures are used to control changes that are performed on a regular basis (i.e. repetitive maintenance or repetitive repair) and would benefit from a more specifically detailed process. The criteria for use in developing new pre-engineered procedures is in Attachment 1 of CC-AA-112. If an approved pre-engineered procedure is not available for controlling a specific temporary change, then a TCCP is required. Activities controlled by pre-engineered procedures are therefore considered as "Exclusions".

Each station in Exelon may have pre-engineered procedures in place that are not available at other stations. Additionally, this procedure (CC-AA-112) identifies other Exclusions that have been agreed upon by all stations as activities that can be implemented without TCCPs. These Exclusions are listed in this Attachment. Various temporary changes are identified as Exclusions based on the simplicity of the change, and commonly acknowledged industry practices associated with performing day to day activities within the plant that do not have an impact on plant design based configuration.

Based on the above, the following table is provided to identify activities that typically require a TCCP, and a list of activities that are typically addressed by pre-engineered procedures. The actual determination of whether or not a specific activity can be performed as a TCCP or a pre-engineered activity depends upon what has been specifically approved for use at individual stations.

Controlled and Issued as TCCPs	Pre-Engineered Activities (See Note 1)
Temporary Setpoint Changes	Ventilation Dampers out of Normal Position (through Operations abnormal lineup procedure)
Mechanical jumpers (hose, tube, pipe) used as pressurized process flowpaths **(CM 6.1.6.3)**	Temp Lead Shielding
Valve Blocks Not Installed Within an Operations Clearance Boundary	Plant Barriers – includes Fire, Ventilation, Security, Radiation, Flood, High Energy Line Break, and Missile Barriers
Temp Power Feeds (TCCP unless Exclusion Item 6 applies)	Scaffolding mounted or attached to structures
Floor Drains with plugs installed	Procedure CC-AA-404 "Maintenance Specification: Application Selection, Evaluation and Control of Temporary Leak Repairs".
Pipe Supports	**Freeze Seals (CM-6.1.3.3)**
Lifted Leads / Pulled Circuit Boards **(CM 6.1.6.3)**	Rigging
Installed or Removed Filters or Strainer	
Gagged or Disabled Relief Valves **(CM 6.1.6.1)**	
Electrical Jumpers (is Maintenance developing a Maint. Alter. Procedure?) **(CM 6.1.6.3)**	
Disabled Alarm	
Battery Cell Jumpers **(CM 6.1.6.3)**	

ATTACHMENT 2
TCCPs, Exclusions and Associated Administrative Controls (CM 6.1.2.1)
Page 2 of 3

Controlled and Issued as TCCPs	Pre-Engineered Activities (See Note 1)
Temp Heat/Cooling for supplementing equipment heating or cooling requirements	
Scaffolding attached to plant system components or appurtenances	
Line Stops	

Note 1: The temporary changes identified in the "Controlled as Procedural Temporary Changes" may not apply to both Regions (Mid-West, and Mid Atlantic). Confirm the applicability of procedures that address these topics using the site Controlled Documents module.

Exclusions and associated Administrative Control Requirements

1. **Surveillance and Inservice tests** are repetitive in nature and typically controlled through specific station procedures which call for temporary configuration change (i.e., installation of a jumper to conduct a trip and cal test, would not fall under this procedure).
2. If evolution of a permanent modification includes **temporary changes required to support the implementation of the permanent modification**, and has been evaluated as part of permanent modification process, then temporary changes are exempted.
3. **Maintenance activities, replacements, troubleshooting and surveillance functions** that are conducted in accordance with an approved procedure, or Work Orders developed from the requirements of task specific station approved procedures. The physical plant configuration must be within the approved design requirements upon exit from the maintenance activity, replacement, troubleshooting or surveillance, or a TCCP is required to consider the SSC as operable.
4. **SSCs included <u>within</u> an Operations Clearance.**
5. **M&TE equipment** discussed in 5.a and 5.b, below, shall be tagged per station procedures for implementing the change. The Work Order number used for installing the M&TE shall be entered into the TCCP Tracking Log for Operator awareness. Additionally, the M&TE items shall be tracked in the TCCP monthly report for use in periodically review by the SM. **(CM-6.1.5.11)**

   a. A TCCP is not required for Measurement and Test Equipment (M&TE) installed on equipment with engineered test points that meet the following requirements:
      - M&TE does not change the system's design function
      - The system is returned to normal configuration before the end of the current refuel cycle.

   b. A TCCP is not required for M&TE installed for troubleshooting efforts on equipment without engineered test points that meet the following requirements: **(CM-6.1.2.7)**
      - M&TE does not change the system's design function
      - M&TE are installed and controlled in accordance with an approved procedure or work package instructions provided that the temporary change of the equipment is clearly documented.
      - The system is returned to normal configuration 90 days after installation. (based on Reference 6.5 and Reference 6.12)
      - Risk significance has been assessed in accordance with Reference 6.9.

Temporary Change Tracking Log

04-00845 NR-V-4A/B Inflatable Plug
Status : Installed

TO BE FILLED OUT BY ENGINEERING:
Title / Description: **NR-V-4A/B Inflatable Plug**

ECR # **04-00845**
W/O or AR #

☐ **MR90 TCCP** (Only check if a 50.59 Screening was NOT performed)

Component Tag #:	**Plant System Number:**	**System Engineer:**
NR-V-0004A	531 Nuclear Services River Water System	Joseph R Bashista

Requesting Department : **Maintenance**
Responsible Design Engineer : **Ronald L Summers/TMI**
Modification restrictions / Compensitory Measures : **Ops to monitor air pressure 1/day (range 100-180 psig). If outside this range contact Shift Maintenane**

Expected Removal Date : **06/30/2005**
Document extension of removal date in the comments field.
Comments:

TO BE FILLED OUT BY OPERATIONS:
Authorized By Operations: Date and Time: **11/18/2004 03:52 PM**

Authorized By: **David B Wilson/TMI**

Make Entry

Installed: Date and Time: **11/18/2004 03:52 PM**

Installed By: **David B Wilson/TMI**

Lifted Leads and Jumpers issued (Engraved Tags) :

Removed: Date and Time:

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Comments:

Form ES-401-5 **Written Exam Question Worksheet** Q # 098

Examination Outline Cross-Reference Tier # 3

Evolution/System Emergency Procedures/Plan Group #

K/A # 2.4.16 Page # 2-12 RO/SRO Importance Rating 3.0 4.0

Measurement Knowledge of EOP implementation hierarchy and coordination with other support procedures.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ✓ 55.41 .10 ✓ 55.43 .5

Proposed Question ☐ RO ✓ SRO ✓ PRA Related **Correct Answer** D.

Plant conditions:

- Control room staff is limited to MINIMUM STAFFING requirements.
- Reactor tripped from 100% power.
- Loss of ICS/NNI HAND Power due to breaker trip.
- Primary-to-secondary leakage: OTSG 1A = 0.7 gpm, OTSG 1B = 0.4 gpm.
- Core exit thermocouple temperatures are 540 degrees F.
- RCS pressure is 950 psig.
- RB pressure is 2.5 psig.
- Loss of 230 KV lines 1091 AND 1092, due to fault at Middletown Junction Substation.

Based on these conditions at the time of trip, from the list below identify the FIRST (highest priority) PROCEDURE to be implemented, when reactor trip immediate actions have been completed.

A. OP-TM-EOP-005, OTSG Tube Leakage.

B. MAP NN-1-3, 230 KV Substation Trouble.

C. 1202-41, Total or Partial Loss of ICS/NNI Hand Power.

D. OP-TM-EOP-002, Loss of 25 Degrees Subcooled Margin.

Technical Reference OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 4.1.5, Priority of EOP Symptoms, Section 4.1.7, Performing Parallel Actions, Pages 8 and 10, Rev. 10.
OP-TM-EOP-001, Reactor Trip, Step 3.1.1, Page 3, Rev. 5.

Open Exam Reference None.

Learning Objective V.E.14.01

Question Source ☐ New ✓ Bank Question # SR5E13-14-Q02

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ✓ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because loss of subcooled margin is a higher priority procedure than the OTSG tube leak.

Distracter is plausible because the entry requirements for OP-TM-EOP-005 are met with a total OTSG tube leakage of greater than 1 gpm.

B INCORRECT because loss of subcooled margin is a higher priority procedure than the MAP alarm.

Distracter is plausible because loss of 1091 and 1092 lines would actuate the 230 KV Substation trouble alarm.

C INCORRECT because loss of subcooled margin is a higher priority procedure than the EP.

Distracter is plausible because 1202-41 Total or Partial Loss of ICS Hand Power is the correct procedure for

Loss of ICS/NNI Hand Power.

D CORRECT. Loss of Sub-Cooling Margin is the highest priority symptom given in the stem IAW OS-24.

Comments None.

4.1.4 Immediate Manual Actions

A. Immediate Manual Actions are high priority actions that are memorized by licensed operators. These steps represent a predetermined prioritization of actions performed before there is opportunity to refer to the appropriate procedure.

B. When a reactor operator recognizes entry into an EVENT PROCEDURE, CRS announces "Entering (procedure name)" and directs "Perform the immediate manual actions"

C. The reactor operator verbalizes and performs the Immediate Manual Actions.

D. The Control Room Supervisor performs verification and signs or otherwise marks completion of Immediate Manual Actions in all EVENT PROCEDURES.

4.1.5 Priority of EOP Symptoms

A. EOP(s) are used to mitigate symptoms that result from transients or other upsets in reactor heat transfer with the following priority:

1. Loss of Subcooling Margin
2. Excessive Primary-to-Secondary Heat Transfer
3. Lack of Primary-to-Secondary Heat Transfer
4. Steam Generator Tube Leak

B. Symptoms are continuously monitored after entry into <u>any</u> EOP.

4.1.7 Performing Parallel Procedures

A. Any other procedure actions should be interrupted to perform Reactor Trip Immediate Manual Actions.

B. Once Immediate Manual Actions have been accomplished, if the use of multiple procedures is required, the Control Room Supervisor is responsible to manage resources and determine the action most significant to overall event mitigation. The CRS determines the sequence of actions if multiple procedures have been initiated.

C. Generally, performance of EOP actions is higher priority than performance of AOP/AP/EP actions and EOP Rules are higher priority than EOP Guides. However when multiple procedures apply, the CRS determines the sequence between these parallel procedure actions in order to perform the actions which are most critical to event mitigation.

D. When an event occurs, the EVENT PROCEDURE directs operation of plant equipment. Other procedures may contain guidance for operation of plant equipment, however, EVENT PROCEDURE actions override guidance in other procedures.

E. When direction from Rules, Guides or procedures conflict, the following order of precedence should be applied: (1) Rules (including the order of priority within the Rules) (2) EOP steps (3) Guides and (4) other procedure requirements.

3.0 VITAL SYSTEM STATUS VERIFICATION (VSSV)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3.1 **IAAT** a symptom exists**, then** immediately treat the symptom using the following priority: 1. SCM < 25°F **GO TO** OP-TM-EOP-002. 2. XHT **GO TO** OP-TM-EOP-003. 3. LOHT **GO TO** OP-TM-EOP-004. 4. OTSG tube leakage > 1 gpm **GO TO** OP-TM-EOP-005.	
_____ Time	
_____ 3.2 **ANNOUNCE** Reactor Trip over plant page and radio (include plant conditions sufficient for NLO response per OS-24).	
_____ 3.3 **VERIFY** control rod groups 1 through 7 are fully inserted.	_____ **INITIATE** Emergency Boration per RULE 5 – EB.
_____ 3.4 **VERIFY** MAIN FW Flow to A & B OTSG are each < 0.5 mlb/hr.	_____ **ENSURE** FW-V-5A AND FW-V-5B are stroking closed or are closed.
_____ 3.5 **VERIFY** OTSG level > setpoint.	_____ **INITIATE** RULE 4 – FWC.
_____ 3.6 **VERIFY** ICS/NNI HAND **or** AUTO Power are available.	_____1. **TRIP** <u>both</u> MFW pumps. _____2. **ENSURE** EFW actuation and **INITIATE** Guide 15. _____3. **CONTROL** OTSG pressure using the ADV B/U loaders _____4. **INITIATE** 1202-40, "Loss of ICS Hand and Auto Power".

Form ES-401-5 **Written Exam Question Worksheet** Q # 099

Examination Outline Cross-Reference Tier # 3

Evolution/System Radiation Control Group #

K/A # 2.3.1 Page # 2-9 RO/SRO Importance Rating 2.6 3.0

Measurement Knowledge of 10 CFR 20 and related facility radiation control requirements.

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

10 CFR Part 55 Content ☑ 55.41 .12 ☑ 55.43 .4

Proposed Question ☐ RO ☑ SRO ☐ PRA Related **Correct Answer** B.

During a primary side tour to check for recurring hot spots, data was recorded to be evaluated for Hot Spot posting and prioritization for flushing operations.

From the list below identify the ONE selection that requires this type of posting and reporting for radiation source elimination.

A. WDL-V-160, inlet to 'B' Precoat Filter:
- Contact dose is 90 mRem/hour.
- Dose rate at 30 cm is 12 mRem/hour.

B. WDL-V-55, MWST inlet from vent header to RCBT:
- Contact dose is 105 mRem/hour.
- Dose rate at 30 cm is 16 mRem/hour.

C. WDL-V-376, WDL-V-304 isolation/test and drain:
- Contact dose is 110 mRem/hour.
- Dose rate at 30 cm is 30 mRem/hour.

D. Low point piping downstream of WDL-V-240, MWST inlet header:
- Contact dose is 120 mRem/hour.
- Dose rate at 30 cm is 35 mRem/hour.

Technical Reference RP-AA-550-1001, Hot Spot and Radiation Source Component Tracking, Section 2.1, Hot Spot, Page 1, Rev. 1.

Open Exam Reference None.

Learning Objective III.F.02.03

Question Source ☐ New ☑ Bank Question # 2003 Audit Q-027

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A INCORRECT because contact reading is less than 100 MR/hour, and 30 cm reading is less than 5 times background.

B Correct answer. Contact reading exceeds 100 MR/hour, and 30 cm reading is greater than 5 times background.

C Incorrect answer. Contact reading exceeds 100 MR/hour, but 30 cm reading is less than 5 times background.

D Incorrect answer. Contact reading exceeds 100 MR/hour, but 30 cm reading is less than 5 times background.

Comments None.

4.1.3. Prerequisites

1. Radiological surveys performed prior to the posting of radiological signs may include:
   - General area and contact dose rates in the area.
   - Smearable contamination levels.
   - Airborne radioactivity levels.

2. Radiological signs shall have a three bladed symbol in magenta, purple or black (black is the least preferred) on a yellow background in accordance with 10 CFR 20.

4.2. <u>Postings for Access to RCAs</u>

4.2.1. For entries to RCAs (e.g., power block, radwaste, etc.), the following is the minimum level of posting: "Caution - Radioactive Material."

4.2.2. For some locations based upon radiological conditions, "Caution - Radiation Area" may be used in lieu of (or in combination with) the "Caution - Radioactive Material" posting specified above.

4.2.3. Additional information may be required on the minimum posting based on current survey information or at the discretion of the Radiation Protection Department (e.g., "RWP Required for Entry," "No Eating, Drinking or Smoking Permitted," etc.).

4.3. <u>Postings within the RCA</u>

4.3.1. General Provisions for Postings

1. Areas shall be conspicuously posted so as to warn personnel approaching the area from any direction.

2. Radiological postings should reflect the radiological condition of an area. If an area to be posted is within a larger, already posted area, then the smaller area's posting should **<u>not</u>** duplicate information contained on the surrounding area's existing posting. An example of this criterion is "roping off" and posting a pump as a contaminated area inside a room that is already posted as a Radiation Area. The posting of "Radiation Area" would **<u>not</u>** be required on the pump posting.

4.3.2. "Caution - Radioactive Material" – Posting used for an area or room in which there is used or stored an amount of licensed radioactive material exceeding ten times the quantity of such material specified in 10 CFR 20 Appendix C.

4.3.3. "Caution - Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

4.3.4. "Caution - High Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent rate in excess of 100 mrem/hr at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

4.3.5. "Caution - Locked High Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent rate greater than or equal to 1000 mrem/hr at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

4.3.6. "Grave Danger - Very High Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from a radiation source or one meter from any surface that the radiation penetrates.

4.3.7. "Neutron Radiation Area" – Area, accessible to individuals, in which radiation levels could result in an individual receiving deep dose equivalent from neutron radiation greater than or equal to 2.5 mrem in one hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

4.3.8. "Caution - Contaminated Area" – Posting used for an area that has smearable contamination present at levels greater than or equal to 1000 dpm/100 cm^2 beta/gamma or 20 dpm/100cm^2 alpha.

4.3.9. "Red Zone" – Posting insert used for an area that is controlled due to the presence (or concern) of discrete radioactive particles greater than a level of 500,000 dpm. This posting insert is normally used in combination with a "Caution - Contaminated Area" posting.

4.3.10. "Yellow Zone" – Posting insert used for the area that surrounds a "Red Zone" at the egress of a "Red Zone" to control the migration of discrete radioactive particles from a Red Zone area to a non-red zone area. This posting insert is normally used in combination with a "Caution - Contaminated Area" posting.

4.3.11. "Caution - Airborne Radioactivity Area"

1. Posting used for a room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations in excess of 0.3 DAC, for derived air concentration values specified in 10 CFR 20 Appendix B.

Form ES-401-5 **Written Exam Question Worksheet** Q # 100

Examination Outline Cross-Reference Tier # 3

Evolution/System Emergency Procedures/Plan Group #

K/A # 2.4.4 Page # 2-11 RO/SRO Importance Rating 4.0 4.3

Measurement Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

10CFR55.43(b)(5). Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .5

Proposed Question ☐ RO ☑ SRO ☑ PRA Related **Correct Answer** A.

Initial plant conditions:
- Reactor tripped from 100% power.
- OP-TM-EOP-001 immediate actions are complete.

Sequence of events:
- Loss of offsite power (LOOP), with EG-Y-1A start failure.
- EF-P-1 tripped.
- Major Main Steam system leak inside the RB.
- ESAS actuation.
- Total EFW flow indication is 510 gpm.
- Current Parameters

Current Parameters	Value	Trend
RCS Temperature	515°F	Lowering
RCS Pressure	1600 psig	Lowering
Pressurizer Level	30 inches	Lowering
RB Pressure	1.5 psig	Rising
OTSG 1A Pressure	850 psig	Lowering
OTSG 1B Pressure	415 psig	Lowering
OTSG 1A Level	5%	Rising
OTSG 1B Level	6 inches	Lowering

Based on these conditions, identify the ONE selection below that describes the HIGHEST PRIORITY action(s) to be performed by the Control Room team and identify the applicable section of EOP-10.

A. Isolate the affected OTSG using Rule 3, XHT.

B. Reduce RCS subcooled margin using Rule 6, PTS.

C. Raise Pressurizer level using Guide 9, RCS Inventory Control.

D. Reduce EFW flow using Guide 15, EFW Actuation Response.

Technical Reference OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 3.6, Excessive Primary-to-Secondary Heat Transfer (XHT), Page 4, Rev. 10. OP-TM-EOP-003, Excessive Primary-to Secondary Heat Transfer, Step 2.A, Page 1, Rev. 2.

Open Exam Reference None.

Learning Objective V.E.21.01

Question Source ☐ New ☑ Bank Question # QR5E21-01-Q07

☐ Modified Bank Parent Question #

Question NRC Exam History

Question Cognitive Level ☐ Memory/Fundamental Knowledge ☑ Comprehension/Analysis

Discriminant Validity Statements

A CORRECT. The highest priority symptom is excessive heat transfer due the main steam system leak.

B INCORRECT because EOP-010, Rule 6 is not the highest priority rule IAW OS-24.

Distracter is plausible because Rule 6 is applicable because no RCPs are running and HPI is running.

C INCORRECT because EOP-010, Guide 9 is not the highest priority rule IAW OS-24.

Distracter is plausible because Guide 9 is the second immediate Manual Action for EOP-003, Excessive heat Transfer.

D INCORRECT because EFW flow does not require to be throttled IAW EOP-010, Guide 15 is not applicable.

Distracter is plausible because EOP-010, Guide 15 does require throttling EFW if only 1 motor driven feed pump is running (given in the stem) and EFW flow is > 515 gpm.

Comments Added Rules, Guides to the original bank question to improve alignment with the generic KA.

3.6 EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER (XHT):

XHT is undesired heat removal by one or both OTSGs. XHT can be confirmed if ALL of the following conditions exist:

- RCS average temperature below 540°F
- Uncontrolled lowering of RCS temperature
- Tsat for OTSG pressure is less than Tcold on affected OTSG(s)

3.7 FEEDWATER:

A water source to the OTSG(s) from either the Main or Emergency Feedwater Systems.

3.8 LACK OF PRIMARY-TO-SECONDARY HEAT TRANSFER (LOHT):

LOHT is the inability of either OTSG to remove sensible heat from the RCS. LOHT can be confirmed if one of the following sets of conditions exists:

- Core exit temperatures rising above 580°F **and** at least one RC Pump operating
- Core exit temperatures rising **and** NO FEEDWATER available
- Core exit temperatures rising **and** RCS circulation can not be confirmed

3.9 MINIMIZE SCM:

An intentional reduction of the reactor coolant pressure temperature relationship as close as practical to the 25°F subcooling margin or RCP NPSH limit. Actions to minimize SCM are described in Guide 8.

3.10 OTSG AVAILABLE:

A physical condition where the OTSG demonstrates level and pressure control. It means the OTSG is in a condition where primary to secondary heat transfer would be possible. Primary to secondary heat transfer need not be demonstrated to determine this availability.

- Primary to secondary leakage should not be considered a means of OTSG level control.
- A dry OTSG is not available.
- An OTSG isolated IAW EOP-005 isolation criteria is not available.

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EXCESSIVE PRIMARY-TO-SECONDARY HEAT TRANSFER

1.0 ENTRY CONDITIONS - Excessive Primary to Secondary Heat Transfer (PSHT) while shutdown prior to DHR operation.

2.0 IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 2.A **PERFORM** Rule 3, XHT.	
___ 2.B **INITIATE** Guide 9, "RCS Inventory Control".	

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.1 **ENSURE** announcement of reactor trip over the plant page and radio.	
___ 3.2 **VERIFY** at least one OTSG has stable pressure with level present.	___ **GO TO** OP-TM-EOP-009.
___ 3.3 **PERFORM** Guide 12, to limit RCS heatup and pressurization.	
___ 3.4 **ENSURE** RCS temperature reduction has been terminated.	___ **If** PSHT is not excessive **and** temperature reduction is due to HPI/Break Cooling, **then GO TO** OP-TM-EOP-006.
___ 3.5 **VERIFY** primary to secondary heat transfer is being established.	___ **GO TO** OP-TM-EOP-004.
___ 3.6 **VERIFY** RCS Tcold > 525°F.	___ **INITIATE** Emergency boration - Rule 5, EB.
___ 3.7 **ENSURE** performance of an alarm review.	