Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K6.04 (003)	
	Importance Rating	2.8	3.1

(K&A Statement) Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation.

Proposed Question: Common 1

Plant conditions:

- Reactor Power 100%
- Intermediate Closed Cooling Water Pump IC-P-1A OOS for maintenance
- MU-P-1A providing makeup and seal injection
- Seal injection flow 20 gpm due to RCP Seal Injection Valve MU-V-32 air leak

Event:

- Inadvertent B Train 30 psig ESAS Containment Isolation actuation occurs

Which ONE of the following events will occur?

A. IC-P-1B will trip.
B. All Reactor Coolant Pumps will trip.
C. RCP Seal Injection will be isolated.
D. Reactor Building Spray Pump 1B starts.

Proposed Answer: B. All Reactor Coolant Pumps will trip.

Explanation (Optional):

A. Plausible since ICCW flow will be lost due to IC-V-3 closing; however IC-V-74 will open to provide Recirc flow in the ICCW system.
B. Correct answer. RCPs will trip if Seal Injection flow is <22 gpm and ICCW flow drops to <550 gpm resulting in a reactor trip, ICCW flow dropped due to 30 psig ESAS.
C. Plausible since RCP Seal Return flow will be isolated; however Seal Injection flow will be unaffected.
D. Plausible since RB Spray Pump 1B does start on a 30 psig RB pressure signal; however a Block 4 permissive signal from the Block loading timers must also be present.

Technical Reference(s): OP-TM-MAP-C0202, IC System Flow Lo (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:			(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

Level 2 – Reference Use

1.0 SETPOINTS

– 550 GPM-Flow switch IC-5-FS

2.0 CAUSES

– IC Pump Failure

3.0 AUTOMATIC ACTIONS

– **If** ICCW flow < 550 gpm for 10 seconds
and seal injection flow is < 22 gpm,
then all RC pumps will trip.

– Standby IC pump starts on Lo flow of 550 gpm.

4.0 MANUAL ACTIONS REQUIRED

4.1 **ENSURE** standby IC pump is Operating.

4.2 **If** IC flow is < 550 gpm,
then INTIATE EP 1202-17, Loss of IC Cooling System.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K1.10 (004)	
	Importance Rating	2.7	2.9

(K&A Statement) Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following systems: CRDS operation in automatic mode control.

Proposed Question: Common 2

Initial plant conditions:

- Reactor at 100% power
- Control Rod index 292%

A transient occurs yielding the following plant conditions:

- Reactor power 100%
- Both Thot instruments 599 °F and lowering
- Control rod index 300%

Which ONE of the following events would cause the above changes in plant conditions?

A. Tave instrument slowly fails high.
B. A Control Rod Drive Motor Fault has caused a continuous rod withdrawal.
C. Adding a batch to the Makeup Tank that has a higher boron concentration than expected.
D. Placing a Deborating Demineralizer in service with a lower than expected Decontamination Factor (DF).

Proposed Answer: C. Adding a batch to the Makeup Tank that has a higher boron concentration than expected.

Explanation (Optional):

A. Plausible if examinee believes rods will move out on this failure.
B. Plausible since this will result in a rod index of 300%; however Tave will not be lowering.
C. Correct – rods move out to compensate for negative reactivity from boron.
D. Plausible if the examinee does not know a lower DF will result in less boron being removed than expected.

Technical Reference(s): 1102-4 pages 49 & 52 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: 642-GLO-10 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.1
55.43 ______

Comments:

Figure 6
Volume of 15000 ppm Borated Water for 5% Rod Withdrawal - Cycle 16
(Group 7 between 30 and 90% WD)

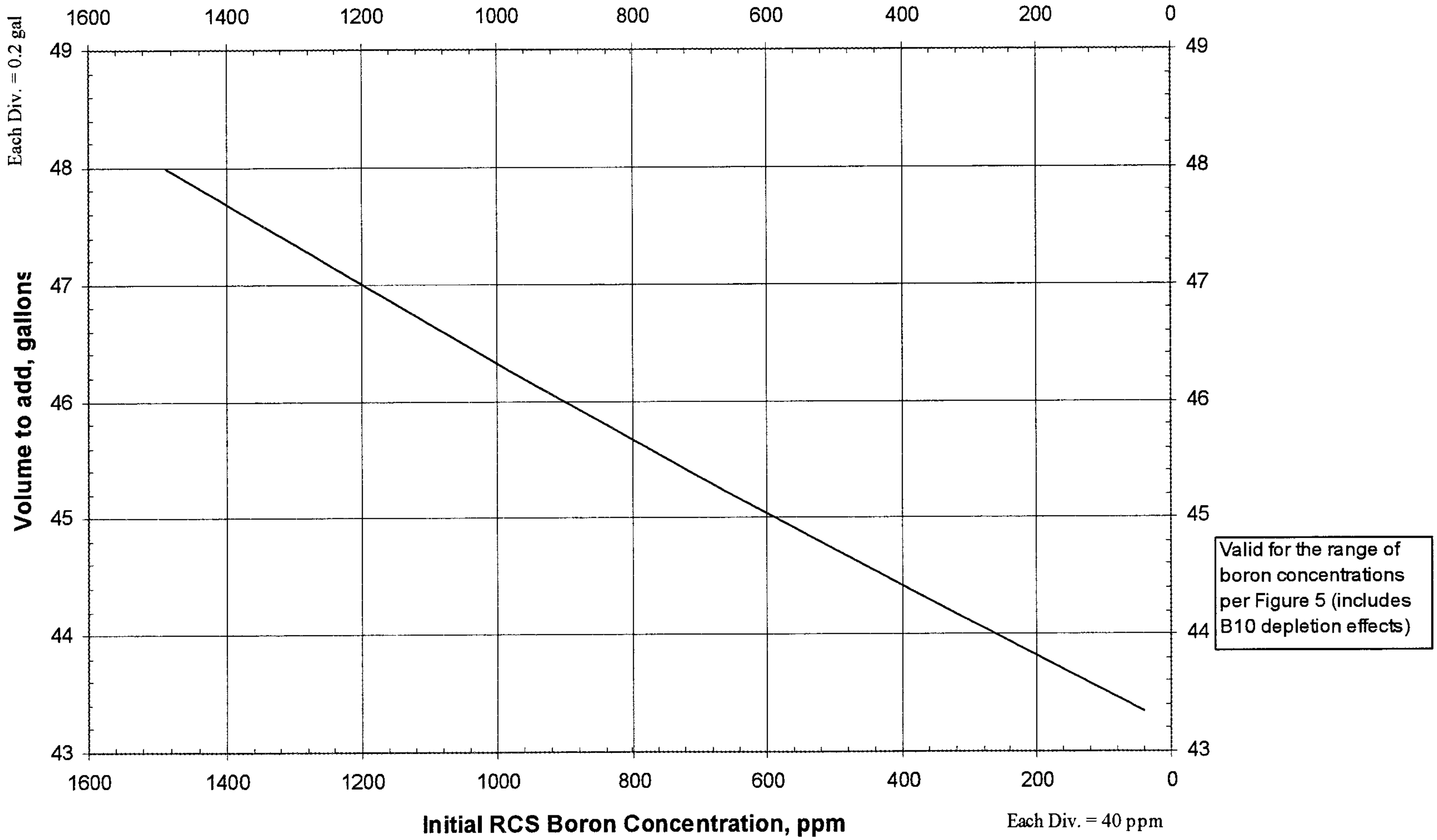
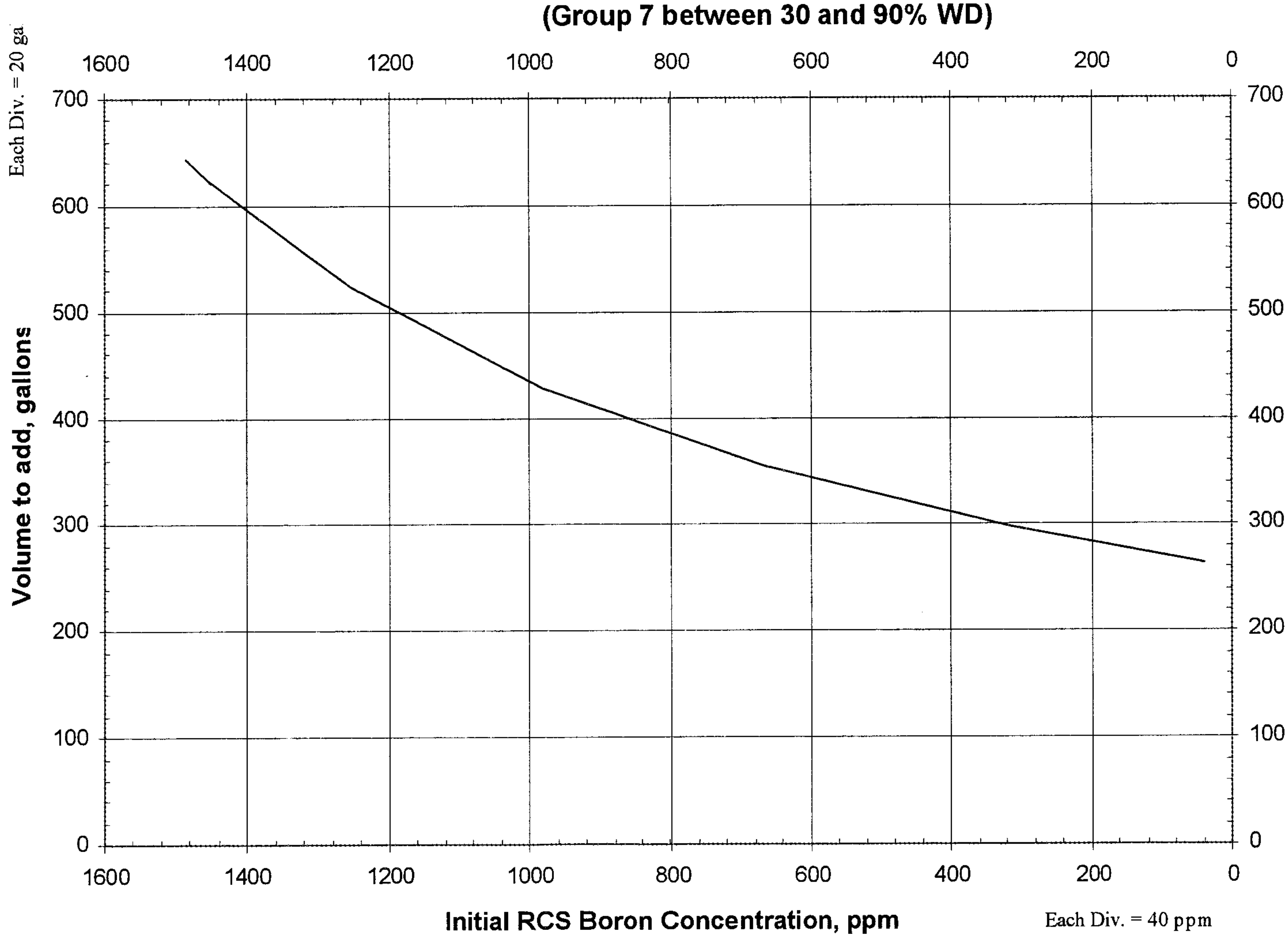


Figure 3
Volume of 2500 ppm Borated Water for 5% Rod Withdrawal - Cycle 16
(Group 7 between 30 and 90% WD)



Valid for the range of boron concentrations per Figure 5 (includes B10 depletion effects)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	K6.03 (005)	
Importance Rating	2.5	2.6

(K&A Statement) Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger.

Proposed Question: Common 3

Which ONE of the following events would prevent Simultaneous DHR and LPI/HPI Operation initiation in accordance with OP-TM-212-921, "Simultaneous DHR and LPI/HPI Operation" ,following an RCS LOCA?

A. Loss of Instrument Air.
B. Loss of Vital bus "B" (VBB).
C. All Reactor Coolant Pumps tripped.
D. Manual isolation of DH Removal Heat Exchanger "B" (DH-C-1B) due to RM-L-3 HI alarm.

Proposed Answer: D. Manual isolation of DH Removal Heat Exchanger "B" (DH-C-1B) due to RM-L-3 HI alarm.

Explanation (Optional):

A. Plausible since DC-V-2B and DC-V-65B valves will fail on loss of instrument air; however they fail to their ESAS positions.
B. Plausible since DC-V-2B and DC-V-65B valves will fail on loss of Vital bus "B" however they fail in their ESAS positions.
C. Plausible since different RCS pressure values will be in effect if an RCP is operating; however the procedure can be initiated in either case.
D. Correct answer. The B Decay Heat Cooler will have to be isolated and OP-TM-212-921 requires both trains of LPI to be operable or operating in the ESAS mode.

Technical Reference(s): MAP Alarm C-1-1, RM-L-3 (DHCCW LOOP B) (Attach if not previously provided)

OP-TM-212-921, Simultaneous DHR and LPI/HPI Operation Page 1

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ______

	Modified Bank #	______ (Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.10
	55.43	______

Comments:

SIMULTANEOUS DHR AND LPI/HPI OPERATION

1.0 PURPOSE

This procedure provides direction for use of Decay Heat Removal cooling mode in the case of a RCS leak which requires emergency injection but the energy removed by the leak is insufficient to remove core decay heat. If Pressurizer level is being maintained without HPI (LPI), then use OP-TM-212-901.

This procedure is appropriate for RCS leaks where DHR is desired for core cooling because the RCS leak size does not allow sufficient LPI or HPI flow to further reduce RCS temperature. This procedure is appropriate with large RCS leakrates (> 400 gpm) if the leak can be isolated (e.g. HPI PORV Cooling or OTSG Tube Rupture). If the RCS leakrate is > 400 gpm, and cannot be isolated or only one LPI train is available, then use LPI & HPI in RB sump recirculation mode per EOP.

2.0 MATERIAL AND SPECIAL EQUIPMENT - None

3.0 PRECAUTIONS, LIMITATIONS, AND PREREQUISITES

3.1. Precautions

3.1.1. Do not cross tie DH trains. Ensure DH-V-38A, DH-V38B, DH-V-12A, and DH-V-7B remain closed.

3.2. Limitations

3.2.1. Maintain pressurizer level > 20" and SCM > 25ºF.

3.2.2. Maintain RCS pressure < 370 psig (or Fig 1A limit if RCPs are operating).

3.2.3. Do not exceed 200 F DH Cooler ΔT, as measured between the inlet shell side (A0109) and the inlet tube side (DH6-TI-2 on CC).

3.2.4. When the DHR system is in operation without any RCP's operating, indicated DH return temperature (DH2-TI-2 on CC) shall be used as the RCS temperature on Fig. 1A (OP-TM-EOP-010) and for Cooldown rate monitoring.

C-1-1
Revision 38

ALARM:

RM-L-3 (DHCCW SYSTEM LOOP B)

SET POINTS:

Refer to Operating Procedure 1101-2.1, RMS setpoints.

CAUSES:

Primary coolant leakage to the DHCCW system Loop "B"

AUTOMATIC ACTION:

None

OBSERVATION (CONTROL ROOM):

1. RM-L-3 "Alert" on PRF
2. RM-L-3 "Hi Alarm" on PRF
3. RM-L-3 Indication on PRF > above setpoints

MANUAL ACTION REQUIRED:

Refer to 1202-12, Excessive Radiation Levels

Alert Hi Alarm:

1. Take sample at DC-V-27B on DHCCW system Loop "B".
2. Have Rad Con perform analysis to verify alarm.

Hi Alarm as verified by sample analysis:

1. Switch Decay Heat Removal to Loop "A" (refer to 1202-35, Loss of DH Removal).
2. Isolate RC to "B" Removal cooler by closing DH suction split isolation valve DH-V-12B.
3. Isolate DHCCW to DHR cooler "B" by closing the inlet and outlet isolation valves DC-V-2B and 3B respectively.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	A4.05 (006)	
Importance Rating	3.9	3.8

(K&A Statement) Ability to manually operate and/or monitor in the control room: Transfer of ECCS flow paths prior to recirculation.

Proposed Question: Common 4

Plant conditions:

- Reactor Coolant System unisolable LOCA in progress
- Decay Heat Pump DH-P-1A flowrate 3150 gpm
- Decay Heat Pump DH-P-1B flowrate 3240 gpm
- MAP E-2-4 BWST Level LO alarm is illuminated
- Reactor Building Flood level is 57 inches
- OP-TM-EOP-010, Guide 21, "RB Sump Recirc", has been initiated

Event:

- DH-V-6A, RB Sump To DH-P-1A Suct Hdr Isol Valve, will not open.
- BWST Level continues to lower

Which ONE of the following is the required action?

A. Place DH-P-1A in Pull-to Lock.
B. Shutdown both Reactor Building Spray Pumps.
C. Initiate OP-TM-212-921, "Simultaneous DHR and LPI/HPI Operation".
D. Open Decay Heat Removal Pump Suction cross-connect valve DH-V-12A.

Proposed Answer: A. Place DH-P-1A in Pull-to Lock.

Explanation (Optional):

A. Correct – RNO action for DH-V-6A failure.
B. Plausible since Reactor Building Spray Pump BS-P-1A will be tripped.
C. Plausible since this procedure would be applicable for leaks >400 gpm; however the leak has to be isolable.
D. Plausible since this would provide a suction cross-connect; however it would remove mechanical separation of the trains and DH-V-12A may not be accessible.

Technical Reference(s): OP-TM-EOP-010, Guide 21 (Step 3) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

Guide 21
Transfer to RB Sump Recirculation

IAAT BWST level < 15 feet **or** RB FLOOD Level > 54 in., **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Maximum RB Flood Level limit (64 in.) is to protect PZR Level and OTSG Level instruments.
Minimum RB Flood Level Limit (32 in.) is to ensure adequate DH & BS pump NPSH.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 1. **ANNOUNCE** initiation of RB Sump Recirculation over the plant page and radio.	
____ 2. **VERIFY** RB FLOOD LEVEL > 32 in.	
3. **When** BWST LEVEL reaches 9.5 ft **or** RB Flood Level > 56 inches, **then OPEN,** ____ DH-V-6A ____ DH-V-6B.	____ 1. **PLACE** the affected train DH pump in PTL. ____ 2. **INITIATE** contingency actions for one DH pump IAW OP-TM-211-901, "Emergency Injection HPI/LPI". ____ 3. **INITIATE** shutdown of affected BS pump IAW Section 5 of OP-TM-214-901, "RB Spray Operation".
4. **When** BWST LEVEL reaches 6.33 ft **or** RB Flood Level > 56 inches, **then CLOSE** the following valves: ____ DH-V-5A ____ DH-V-5B ____ BS-V-2A ____ BS-V-2B	
____ 5. **INITIATE** Guide 22 "RB Sump Recirculation."	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K3.01 (007)	
	Importance Rating	3.3	3.6

(K&A Statement) Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment.

Proposed Question: Common 5

Plant conditions:

- Reactor Coolant System fill in progress
- The Pressurizer is vented to the Reactor Coolant Drain Tank
- All Reactor Coolant Drain Tank manual vent valves are closed
- The Reactor Vessel is vented to the Reactor Building via the CRDM vents
- Hot Legs are vented to the Reactor Building via high point vents

Event:

- WDG-V-4, Containment Isolation RB Vent Header, fails closed

If the fill evolution were to continue, which ONE of the following describes when Reactor Building Sump Level will begin to rise?

A. After the Pressurizer has filled, when the RC Drain Tank overflows through the relief valve.
B. After the Pressurizer has filled, when the RC Drain Tank rupture disk has failed.
C. Before Pressurizer level reaches 400 inches, when flow begins out the CRDM vents.
D. Before Pressurizer level reaches 400 inches, when flow begins out the Hot Leg vents.

Proposed Answer: C. Before Pressurizer level reaches 400 inches, when flow begins out the CRDM vents.

Explanation (Optional):

A. Plausible if the examinee believes the Pressurizer would fill and overflow to the RC Drain Tank causing it to pressurize; however the level of the CRDM vents is below the full point of the Pressurizer.
B. Plausible if the examinee believes the system will pressurize as the fill continues; however the Reactor Vessel is vented to the RB.
C. Correct answer. The Pressurizer and RC Drain Tank will be hydraulically locked which will cause the Reactor Vessel to overflow at a lower indicated pressurizer level than expected.
D. Plausible if the Examinee does not know the level of the Hot Leg Vents is above the CRDM vents but does recognize the manometer affect of RCDT isolated.

Technical Reference(s): Drawing 308-946 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

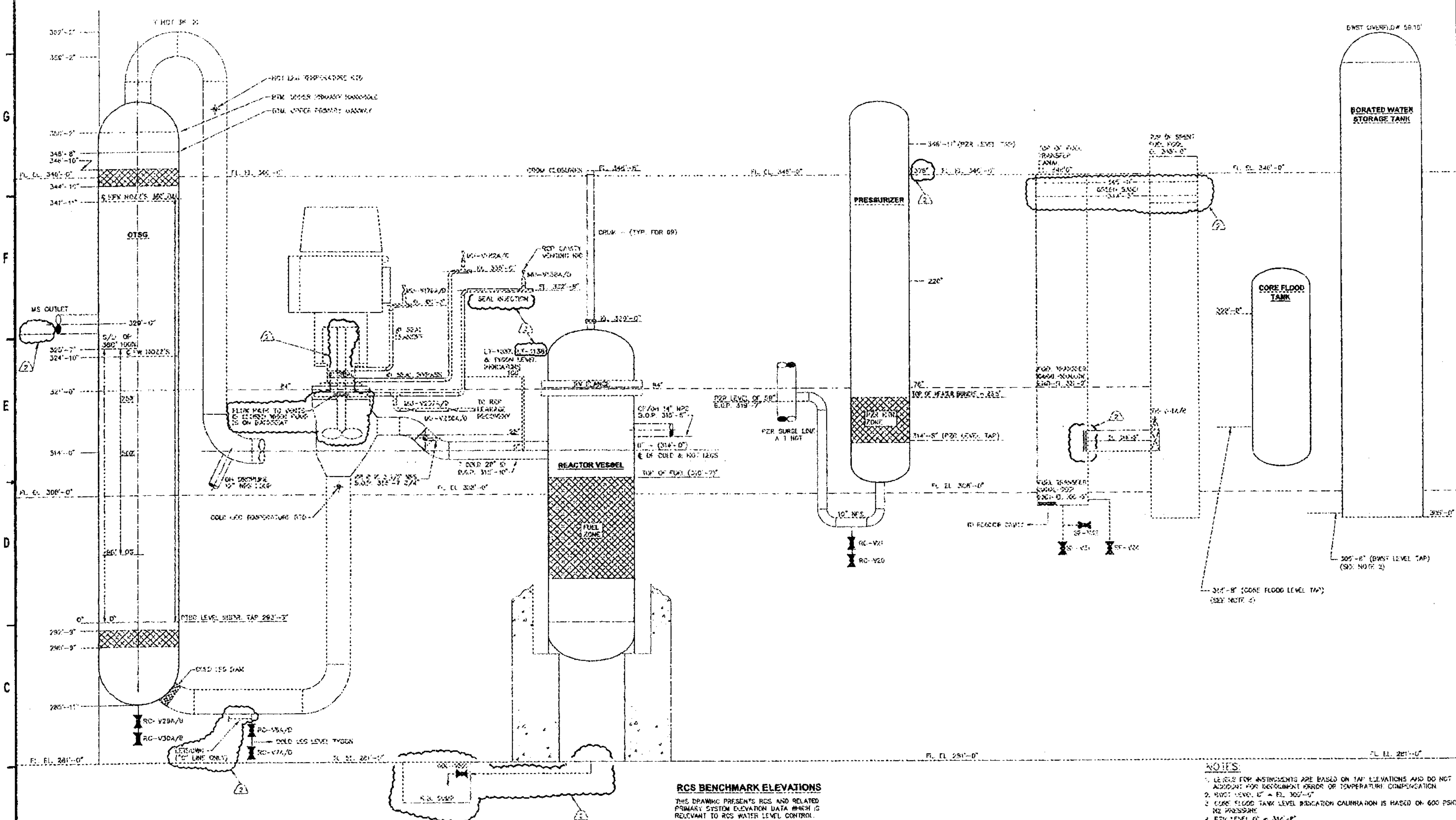
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.3
55.43 ______

Comments:



RCS BENCHMARK ELEVATIONS

THIS DRAWING PRESENTS RCS AND RELATED PRIMARY SYSTEM ELEVATION DATA WHICH IS RELEVANT TO RCS WATER LEVEL CONTROL.
THIS DRAWING CANNOT BE USED FOR ELEVATION INFORMATION NOT PRESENTED ON THE DRAWING.
THIS DRAWING IS NOT DRAWN TO SCALE.
THE SYSTEM CONFIGURATIONS ARE SIMPLIFIED (ONLY ONE OF THE REDUNDANT COMPONENTS IS PRESENTED). SYSTEM FLOW DIAGRAMS (302 SERIES) MUST BE USED TO DETERMINE SYSTEM FLOW PATHS.

NOTES:

1. LEVELS FOR INSTRUMENTS ARE BASED ON TAP ELEVATIONS AND DO NOT ACCOUNT FOR INSTRUMENT ERROR OR TEMPERATURE COMPENSATION.
2. BWST LEVEL 0" = EL. 305'-0"
3. CORE FLOOD TANK LEVEL INDICATION CALIBRATION IS BASED ON 600 PSIG N2 PRESSURE
4. PZR LEVEL 0" = 314'-8"
5. RV DRAINDOWN LEVEL (LT-1037, LT-1138) 0" = 314'-0"
6. BOP = BOTTOM OF PIPE

NO	DWG. NO.	TITLE
8	G/C 304004	M.U. & PURIF. PLAN & SECTIONS
7	G/C 304663	M.U. & PURIF. PLAN, SECT. DETLS
6	B&W 306595	INTERNALS GENERAL ARRANGEMENT
5	B&W 270387	CORE SUPPORT ASSEMBLY
4	B&W 153436	REACTOR CLNT. PIPING ARRGT
3	B&W 142632	ASSY. OF CORE FLOOD TANK
2	B&W 131192	STEAM GEN. LONGITUDINAL SECTION
1	B&W 131100	STEAM GENERATOR OUTLINE

REFERENCES

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

AmerGen

REACTOR COOLANT SYSTEM
RCS BENCHMARK ELEVATIONS
SCHEMATIC LAYOUT

TMI-1 | DWG. NO. 308-946 | SH. N/A | REV. 2
SCALE N/A

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	G2.1.2 (008)	
Importance Rating	3.0	4.0

(K&A Statement) Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: Common 6

To prevent the formation of condensation in the Control Rod Drive (CRD) stator water jacket ....

A. maintain Intermediate Closed Cooler outlet temperature >90°F.
B. backwash the Intermediate Closed coolers if differential pressure is >7 psig.
C. maintain Intermediate Closed Cooling Water filter differential pressure <12 psid.
D. do NOT operate with only one Intermediate Closed cooler in service for >24 hours.

Proposed Answer: A. maintain Intermediate Closed Cooler outlet temperature >90°F.

Explanation (Optional):

A. Correct – per reference Limitation 2.2.10.
B. Plausible since this is an action to be taken; however fouling of the coolers would raise ICCW temperature preventing formation of condensate in the stator water jackets.
C. Plausible since changing out the filters should be accomplished if differential pressure is >12 psig; however higher filter delta P would result in lower ICCW flow and higher temperature, which would help prevent condensate formation.
D. Plausible since this is a caution for system operation; however having only one cooler in operation could result in higher ICCW temperature which would help prevent condensate formation.

Technical Reference(s): OP-TM-541-000, Primary Component Cooling (Page 4) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

Comments:

2.2.4 To ensure heat removal capability remains available, backwash NS cooler if river water cooler (tube side) DP > 7 psid.

2.2.5 To avoid clogging of the NR strainers (i.e. keep strainer pressure >20 psig), do not operate for extended periods (> 4HRs) with NR-PI-217 pressure in the restricted region on Attachment 7.2.

2.2.6 To prevent excessive pump wear or damage, do not operate NR pumps for extended periods (> 4HRs) with NR-PI-217 pressure in the restricted region on Attachment 7.2.

Nuclear Service Closed Cooing Water

2.2.7 To avoid overheating of components cooled by NS, do **not** exceed 95°F as indicated on (PPC Point A0330) Nuclear Service Closed Heat Exchangers Outlet temperature (Normal band - 70°F and 95°F on PPC Point A0330).

2.2.8 To avoid overpressurizing the NS system, do **not** exceed 45 psig as read on NS-PI-334.

Intermediate Closed Cooling Water

2.2.9 To avoid filter damage, do **not** allow IC-F-1A or IC-F-1B (CRDM cooling water) ΔP to exceed 12 PSID as indicated by IC-11 DPI (local).

2.2.10 To minimize the possibility of forming condensate in the CRD stator water jacket **and** provide adequate CRD stator cooling, IC outlet temperature should be maintained between 90°F and 100°F on IC-6TI (CR).

2.2.11 To prevent pump overheating, maintain flow $\geq$ 40 gpm per pump by IC-5-FI (CR).

2.2.12 To motor overload and potential damage, do **not** exceed IC pump flow of 1050 gpm.

2.2.13 To avoid IC cooler damage do **not** continue operating with only one IC cooler in service for > 24 hours.

2.2.14 To avoid loss of efficiency, if > 7 psid across the IC-C-1s (tube side) backwash the IC coolers.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	A2.03 (010)	
Importance Rating	4.1	4.2

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

Proposed Question: Common 7

The reactor was manually tripped from 100% power due to lowering pressure caused by the Power Operated Relief Valve (PORV) failing open.

The following conditions exist;

- Efforts to close the PORV Block Valve have failed
- RCS pressure 1970 psig and lowering
- Thot is 555 °F and slowly lowering
- Tcold is 545 °F and stable
- Pressurizer level indicates 375 inches and rising

Which ONE of the following actions is the NEXT to be taken?

A. Initiate, "Control Building Ventilation System Radiological Event Operations", in accordance with OP-TM-826-901.
B. Bypass the 1600 PSIG ESAS signal prior to actuation in accordance with OP-TM-642-901, "1600 PSIG ESAS Actuation".
C. Push the 1600 PSIG ESAS Manual PBs to Initiate HPI in accordance with OP-TM-EOP-010, Rule 1, "Loss of Subcooling Margin (SCM)".
D. Place Normal Makeup Valve MU-V-17 in Hand and maintain Subcooling Margin >70 °F in accordance with OP-TM-AOP-043, "Loss of Pressurizer (Solid Ops Cooldown)".

Proposed Answer: D. Place Normal Makeup Valve MU-V-17 in Hand and maintain Subcooling Margin >70 °F in accordance with OP-TM-AOP-043, "Loss of Pressurizer (Solid Ops Cooldown)".

Explanation (Optional):

A. Plausible since this is an IAAT action in EOP-006, LOCA COOLDOWN, but an ESAS actuation has not occurred.
B. Plausible if RCS pressure control is obtained by solid plant operation before reaching 1600 psig, otherwise ESAS actuation would be allowed to occur.
C. Plausible since subcooling margin (SCM) may be lost; however given conditions indicate SCM still exists.
D. Correct – IAAT PZR level > 370" then Section 5.0 requires >70 °F SCM.

Technical Reference(s): OP-TM-AOP-043, Loss of Pressurizer (Solid Ops Cooldown)(Page 9) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

5.0 SOLID PLANT OPERATION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Unit Status Pressurizer conditions require solid operation to control SCM.	
___ 5.1 **VERIFY** reactor shutdown.	___ **TRIP** the reactor.
___ 5.2 **ENSURE** RC-V-3 is OPEN.	
CAUTION Excessive SCM may challenge PORV and Code Safety valves.	
___ 5.3 **PLACE** MU-V-17 in HAND **and RAISE** MU flow to maintain SCM > 70 °F.	
☐ 5.4 **IAAT** SCM approaches 250 °F **then LOWER** MU flow.	
☐ 5.5 **IAAT** all of the following conditions exist: – Valid pressurizer level indication is available, – Adequate pressurizer heaters are available to control RCS pressure, **then GO TO** Guide 8.1, "Recovering a Pressurizer Steam Bubble".	
___ 5.6 **STABILIZE** RCS temperature.	
___ 5.7 **VERIFY** letdown flow > 30 gpm.	___ **INITIATE** OP-TM-211-950, "Restoration of letdown".
___ 5.8 **VERIFY** both trains of DH system are operable.	
___ 5.9 **When** SM determines cooldown is required, **then PERFORM** Brief using Attachment 1, "Solid Operations Cooldown Briefing" as guidance".	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	K6.01(010)	
Importance Rating	2.7	3.1

(K&A Statement) Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems.

Proposed Question: Common 8

The plant is operating at 100% power when the following is observed:

- RC3-PR Hot Leg A Narrow Range Channel 1 fails to 2400 psig
- MAP alarm H-3-2, SASS Mismatch is illuminated
- Console Center Narrow Range Pressure SASS selector switch for RC3A-PT1 Hot Leg A Narrow Range Channel 1 has the red and white lights illuminated
- Console Center Narrow Range Pressure SASS selector switch for RC3B PT1 Hot Leg B Narrow Range Channel has its red light illuminated

Which ONE of the following occurs?

A. The Pressurizer Spray Valve opens.
B. The Power Operated Relief Valve opens.
C. The SCR controlled Pressurizer Heaters energize.
D. No response. SASS has transferred to an alternate channel.

Proposed Answer: A. The Pressurizer Spray Valve opens.

Explanation (Optional):

A. Correct answer. The Spray Valve will open to 40% when indicated pressure is above 2205 psig since the SASS control did not transfer.
B. Plausible since the PORV would open on this type of failure if pressure went above 2450 psig.
C. Plausible if the examinee misinterprets the failure. Pressurizer heaters will fail off in this situation.
D. Plausible if the examinee thinks SASS has operated properly.

Technical Reference(s): OP-TM-MAP-G0308, RC Press Narrow Rng HI/LO page 1 of 2 (Attach if not previously provided)

OP-TM-621-000
ATTACHMENT 7.3 Page 1 of 2

Proposed references to be provided to applicants during examination: None

Learning Objective:	624-GLO-11		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

1.0 SETPOINTS

Hi - >2255 psig or Lo <2055 psig from:

RC3-PR Hot Leg A Narrow Range Channel 1

RC3 PIS Hot Leg B Narrow Range Channel

2.0 CAUSES

- RC-V-1 Pressurizer spray valve Open
- Faulty pressurizer heater operation
- RCS temperature rising on Lowering
- Primary coolant leak
- RC-RV-2 PORV Open
- Failure of controlling RCS Pressure Instrument RC3-PR or RC3-PIS (CC)
- Loss of ICS-Auto or Hand Power

3.0 AUTOMATIC ACTIONS

- Reactor Trip at RCS pressure >2355 psig or <1900 psig.

ATTACHMENT 7.3
SASS Inputs and Indications
Page 1 of 2

SASS		SOURCE INSTRUMENT		INDICATIONS				
Channel No.		A	B	A	B	Non Selected	Selected	Affected ICS/NNI Controls
1-1-1	Reactor Power	NI-5	NI-6	NI-5 Dig. RPS A Ind. A0582	NI-6 Dig. RPS B Ind. A0583		NI-5 NIR A5022	Diamond, Rx Mstr & FW Loop Masters (Neutron Error)
1-1-2	RCS Pressure	RC3A-PT1	RC3B-PT1	RC3A-PR1 RPS A Ind. A0586	RC3B-PR1 RPS B Ind. A0587		A5071	Pressurizer Heaters, Spray Valve & PORV
1-1-3	RCS Flow Loop A	RC14A-DPT1	RC14A-DPT2	RPS A Ind. A0594	RPS B Ind. A0596		RC14A-FI A5069	Stm Gen/Rx Mstr, FW/Valves & FW Pumps (FW Ratioing, BTU Limits & Total Flow ULD Runback
1-1-4	RCS Flow Loop B	RC14B-DPT1	RC14B-DPT2	RPS A Ind. A0595	RPS B Ind. A0597		RC14B-FI A5070	Stm Gen/Rx Mstr, FW/Valves & FW Pumps (FW Ratioing, BTU Limits & Total Flow ULD Runback
1-2-1	T Hot Loop A	RC4A-TE4	RC4A-TE1	A0405	A0508		RC4A-TI A5064	Diamond & Rx Mstr & FW Valves & FW Pumps (T_{avg} Error & BTU Limits)
1-2-2	T Hot Loop B	RC4B-TE4	RC4B-TE1	A0407	A0509		RC4B-TI A5065	Diamond & Rx Mstr & FW Valves & FW Pumps (T_{avg} Error & BTU Limits)
1-2-3	T Cold Loop A	RC5A-TE3	RC5A-TE1	A0511	A0510		RC5A-TI2 A5060	Diamond & Rx Mstr & FW Masters (T_{avg} Error & FW Ratioing)
1-2-4	T Cold Loop B	RC5B-TE3	RC5B-TE1	A0514	A0513		RC5B-TI1 A5062	Diamond, Rx Mstr & FW Masters (T_{avg} Error & FW Ratioing)
1-3-1	F W Temp	* SP5-TE1	* SP5-TE2	A0555	A0398		SP-TI A5054	FW Valves, Pumps & Diamond & Rx Mstr
1-3-2	Spare	* The convention of this SASS channel (1-3-1) is different from all of the other ones in that the "B" source instrument (as opposed to the "A") is the preferred and normally selected ICS/NNI input. The "B" source instrument in this case also is the one that when selected illuminates the red "NORMAL ICS INPUT SELECTED" lamp on the SASS Rack 1 indicator module located just to the right of the power supply module.						
1-3-3	Spare							
1-3-4	Spare							

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	K6.10(012)	
Importance Rating	3.3	3.5

(K&A Statement) Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Permissive circuits.

Proposed Question: Common 9

Plant conditions:

- Reactor power at 100%
- Reactor Protection System (RPS) Channel C tripped on a NI-7 detector failure and has been placed in Manual Bypass.

Event:

- A malfunction within the RPS Channel D Shutdown Bypass bistable causes the channel to go into the Shutdown Bypass mode
- MAP alarm G-3-2, RPS Shutdown Bypass is illuminated

Which ONE of the following will occur?

A. RPS Channel D trips on low pressure.
B. RPS Channel D trips on high pressure.
C. An automatic reactor trip on low pressure.
D. An automatic reactor trip on high pressure.

Proposed Answer: B. RPS Channel D will trip on high pressure

Explanation (Optional):

A. Plausible if the examinee does not know the 1720 psig trip is a high pressure trip, not a low pressure trip.
B. Correct – Shutdown Bypass inserts a 1720 High Pressure Trip.
C. Plausible since RPS Channel D will trip; however Channel C, while tripped, is in Manual Bypass so the reactor will not trip.
D. Plausible because the second part is correct; however Channel C, while tripped, is in Manual Bypass so the reactor will not trip.

Technical Reference(s): OP-TM-MAP-G0302, RPS Shutdown Bypass (Attach if not previously provided)

OP-TM-641-000, Reactor Protection System (RPS/DSS)(Page 2) 2.1.6

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

Level 2 – Reference Use

1.0 **SETPOINTS** - None

2.0 **CAUSES**

- RPS Channel A, B, C or D shutdown bypass switch in bypass position.

3.0 **AUTOMATIC ACTIONS**

3.1 Eliminates the following trips:

- Power/Imbalance/Flow
- Power/pumps
- Low pressure
- Pressure/temperature

3.2 Inserts RCS high-pressure trip setpoint at 1720 psig.

3.3 Illuminates “Manual Bypass” lamp on outside of associated RPS cabinet.

4.0 **MANUAL ACTIONS REQUIRED**

4.1 **If** cause for switch manipulation is unknown, **then INVESTIGATE.**

- **COMPLY** with Tech Specs.
  - **If** reactor is shutdown, **then REFER** to T.S. 2.3.1.f.
  - **If** reactor is at power, **then REFER** to T.S. 3.5.1.4.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

2.1.1 To avoid a spurious DSS Trip, then ensure DSS in defeat if any of the following is to be performed:

- Closing of 1G-2A or 1L-2A breakers
- Work inside the DSS cabinet.
- Work on transmitters RC-PT-949 or RC-PT-963.
- Work inside the A1 or B1 RSD signal conditioning cabinets.

2.1.2 To avoid a common mode failure, upon a detection of a failure that prevents trip action during course of routine surveillance testing at power, all four subassemblies will be tested for that function after which the rotational test cycle will be started again continuing with the same sequence. (reference T.S. 4.1 bases)

2.1.3 To avoid an inadvertent Reactor Trip during maintenance, always place the RPS channel in Manual Bypass before removing modules for repair or replacement. This places the RPS system in a 2 out 3 trip logic.

2.1.4 To avoid a Reactor Trip, do **not** remove a Reactor Trip module unless all the other channels are reset and Diamond is in Manual. Even if the channel is in Manual Bypass, removal of the Reactor Trip module will cause a trip signal to be seen by the other three channels and will trip CRDM breaker(s) associated with that channel.

2.1.5 To avoid a Reactor Trip, do **not** place more than one channel in a tripped or test state.

2.1.6 To avoid a channel trip during power operations, do **not** place a channel in Shutdown Bypass unless it is in Manual Bypass.

2.2 Limitations - None

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	K2.01(013)	
Importance Rating	3.6	3.8

(K&A Statement) Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control.

Proposed Question: Common 10

Which ONE of the choices correctly fills in the blank in the following statement?

Loss of the _________ will prevent the 1A Diesel Generator from supplying 1D 4160 Volt ES Bus if a Loss of Coolant Accident (LOCA) and coincidental Loss of Offsite Power (LOOP) were to occur.

A. VBA 120 Volt Vital Bus
B. 1P 480 Volt Switchgear
C. 1C 480 Volt ES Valves MCC
D. 1A 250 Volt DC Distribution System

Proposed Answer: D. 1A 250 Volt DC Distribution System

Explanation (Optional):

A. Plausible since this bus does provide power to the A ESAS channels; however it will not prevent the diesel from starting and loading onto the 1D 4160 Volt Bus on UV.
B. Plausible since this bus does supply ESAS equipment; however it will not prevent the diesel from operating if it is de-energized.
C. Plausible since this bus does supply ESAS equipment; however it will not prevent the diesel from operating if it is de-energized.
D. Correct – diesel starts but control power to breakers is lost.

Technical Reference(s): EP 1202-9A, Loss of "A" DC Distribution System, 1.0.B Pages 2, 3 and 14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

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.

- Loss of Distribution Panel 1H.
- Loss of Control Power to the following:
  - a. 1A and 1B 6900V Reactor Plant Swgr. Feeder Breakers
  - b. 1D-4160V ES Swgr.
  - c. 1A-4160V Turbine Plant Swgr.
  - d. 1C-4160V Turbine Plant Swgr.
  - e. 1C-480V Turbine Plant Swgr.
  - f. 1E-480V Reactor Bldg. Ht. and Vent Swgr.
  - g. 1G-480V Reactor Plant Swgr.
  - h. 1H-480V Aux. and Fuel Bldg. Ht. and Vent Swgr.
  - i. 1J-480V Turbine Plant Swgr.
  - j. 1N-480V Turbine, Reactor and Control Bldgs. Htg. Swgr.
  - k. 1P-480V ES Swgr.
  - l. 1R-480V Scrn. Hse. ES Swgr.

2.0 **IMMEDIATE ACTION**

A. Automatic Action

1. Suction Valves to VA-P-1A and VA-P-1C close.
2. Emergency Feed Pump EF-P-1 Starts.
3. Emergency Diesel Generator EG-Y-1A starts.
4. Panel 1M automatically transfers to "B" DC Distribution System.
5. RC-RV-2 fails closed and is inoperable.
6. MU-V-6A fails open and MU-V-6B fails closed placing "A" makeup and purification demineralizer in service.
7. MU-V-11B fails open and MU-V-11A fails closed placing MU-F-1B in service.
8. CM-V-1 and CM-V-3 fail closed making RM-A-2 inoperable.

TABLE 3

ES Diesel Gen. A Distribution Panel 1P	Failure Mode
Diesel Gen. A Cranking Control	Starts cranking
Diesel Gen. A Gov. Shutdown and Alarm Control	Inop.
Diesel Gen. A Fuel Pump Aux. Relay Control	Inop.
Diesel Fuel Tank level Switch (DF-LS-152)	Inop.
Diesel Gen. A Exciter Bias	Excitation defeated (generator inoperable)
Diesel Gen. A DC Fuel Pump (DF-P-1B)	Inop.
Diesel Gen. A DC Backup Fuel Transfer Pump (EG-P-10A)	Inop.
Diesel Gen. A Annunciators	Inop.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G2.1.32(013)	
	Importance Rating	3.4	3.8

(K&A Statement) Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: Common 11

Plant conditions:

- Reactor tripped due to a LOCA
- 1600 psig ESAS actuation has occurred
- Decay Heat Pump 1A did not start automatically
- Loss of offsite power (LOOP) occurred following the ESAS actuation
- ESAS signals have **NOT** been cleared

In the above situation.....

A. ES selecting and starting Makeup Pump MU-P-1B on the 1E 4160V Bus could cause the bus to overload.
B. starting Decay Heat Removal Pump 1A on the 1D 4160V bus before block loading is complete will cause diesel generator damage.
C. clearing the disagreement lights before resetting the 27/86 Undervoltage Lockout Relays for busses P, S, R and T will prevent overloading the diesel generator.
D. pressing two out of three 1600 psig BYPASS pushbuttons while block loading is in progress is acceptable as long as no equipment positions are changed.

Proposed Answer: C. clearing the disagreement lights before resetting the 27/86 Undervoltage Lockout Relays for busses P, S, R and T will prevent overloading the diesel generator.

Explanation (Optional):

A. Plausible since starting a second makeup pump on the diesel may overload it; however MU-P-1C would trip when the 43SS is rotated to the MU-P-1B position.
B. Plausible if the examinee thinks they must wait till block loading is over to start the pump; however the diesel is designed to handle the start of the pump, if it did cause and overload protective relaying would protect the generator
C. Correct answer. The disagreement lights need to be cleared to prevent auto-start of some equipment when the lockouts are reset.
D. Plausible if the examinee thinks bypassing the ESAS signal before block loading is complete will not affect equipment positions.

Technical Reference(s): OP 1105-3, Safeguards Actuation System (Pages 4 and 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

1.0 **REFERENCES**

1.1 System Descriptions

1.1.1 E. S. Actuation System - Vol. III, Section F-6, Operations Plant Manual

1.1.2 Class 1E Electrical Systems Vol. I, Section A-1, Operations Plant Manual

1.2 Drawings

1.2.1 GAI-209-451 Thru 649

1.3 Manufacturers Instruction Manuals

1.3.1 Bailey N.I./RPS/ESS Instructions Volume 1, 2 and 3

1.4 FSAR - TMI-1

1.4.1 Section 6 - Engineered Safeguards

1.4.2 Section 7 - Actuation Instrumentation

1.5 Technical Specifications

1.5.1 T/S 3.3 - ECCS

1.5.2 T/S 3.5 - Instrumentation Systems

2.0 **LIMITS AND PRECAUTIONS**

2.1 Equipment

2.1.1 The 27/86 Undervoltage Lockout Relays for Busses P, R, S and T, on Panel "PCR", should not be Reset until all Amber "Disagreement" Lights have been cleared and it is determined that the Diesels can accept the added Load.

2.1.2 Do Not start a 2nd makeup pump on an ES Bus during a blackout with diesels operating. Rotation of the 43SS during this condition will cause the operating makeup pump on the same ES Bus to trip.

2.2 Administrative

2.2.1 During R.C.S. Heatup or Cooldown, if Channels are bypassed and R.C.S. Pressure is below the Trip Bistable Setpoints, Do Not Depress the "Enable and Channel Reset" Pushbuttons as this would initiate E. S.

2.2.2 To Prevent E. S. Actuation during an R.C.S. Heatup, it is essential that the Low Press. Trip Bistables and the Hi Press. Trip Bistables for Each Channel of Actuation A and B be Verified Reset (Return to the Untripped State) prior to increasing > 900 PSI and > 1750 PSI respectively.

2.2.3 To prevent E. S. Actuation during an R.C.S. Cooldown, it is essential that Each Hi. Press. Channel and Each Low Press. Channel of Actuation A and B are "Bypassed" prior to Decreasing < 1650 PSI and < 550 PSI respectively.

2.2.4 Following any ESAS Actuation

- Verify that all components actuate properly and reach the required end state.
- Verification should be accomplished prior to BYPASS or DEFEAT of any actuation signal, except when there is an urgent need to gain control of actuated equipment (e. g., inadvertent actuation or HPI throttling required).
- Once complete actuation has been verified, the actuation signal may be BYPASSED or DEFEATED at any time in preparation to gain control of actuated equipment, when authorized by the Control Room Supervisor/Shift Manager.
- Once complete actuation has been verified and a redundant actuation criteria is met, additional actuation is not required to satisfy the procedure requirement.

3.0 OPERATING PROCEDURES

3.1 System Start-up - Level 1

3.1.1 Prerequisites

_____ 1. Emergency Electrical System is lined up per Operating Procedure 1107-2.

3.1.2 Procedure:

_____ 1. Enclosure 1, System Startup Checklist is complete.

NOTE
4 psig R.B. Press Bistables should not be actuated in this plant condition.

2. Verify RB Pressure is < 2.5 psig and ENABLE/VERIFY ENABLED the following:

	On CC - A Side ESAS:		A Side Manual:
_____	RB1A	_____	PB2/RBA
_____	RB2A	_____	PB3/RBA
_____	RB3A	_____	PB4/RBA

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A3.01(022)	
	Importance Rating	4.1	4.3

(K&A Statement) Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation.

Proposed Question: Common 12

Plant conditions:

- Reactor tripped due to a large RCS Leak in containment (LOCA)
- All RB Emergency Coolers are in operation, per OP-TM-534-901, "RB Emergency Cooling Operations"
- RB Emergency Cooler "A" indicates 32 gpm leakage
- RB Emergency Cooler pressure indicators RR-PI-224, RR-PI-225, RR-PI-226 read 52 psig
- Individual RB Cooler outlet flows read as follows:
  - "A" cooler 2345 gpm
  - "B" cooler 2420 gpm
  - "C" cooler 2120 gpm

Which ONE of the following actions is required concerning continued operation of RB Emergency Coolers?

A. Throttle closed Reactor Building Emergency Cooling Coil Discharge Valve RR-V-6 locally.
B. Secure the 1A RB Emergency Cooling Fan and isolate the respective cooler.
C. Dispatch an operator to close RR-V-10A, RR-P-1A Recirculation Valve.
D. Throttle closed on RB Emergency Cooler "B" outlet valve RR-V-4B.

Proposed Answer: A. Throttle closed Reactor Building Emergency Cooling Coil Discharge Valve RR-V-6 locally.

Explanation (Optional):

A. Correct answer. RR-V-6 should be manually throttled locally to raise cooler outlet pressure to between 55 and 60 psig to ensure maintenance > RB peak pressure.
B. Plausible since this would be the correct action if the leak was > 50 gpm.
C. Plausible since RR-V-10 would be closed if pump flow is insufficient.
D. Plausible since this would reduce flow through the B cooler and raise flow through the other two; however the correct action is to evaluate the C cooler for a leak due to the low flow.

Technical Reference(s): OP-TM-534-901, RB Emergency Cooling Operations (Page 6) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

4.3 Monitor System Operation (After Manual Or ES Actuation)

4.3.1 **ENSURE** NS-V-85 is CLOSED. ___

4.3.2 **IAAT** RB pressure > 2 psig, **then ENSURE** AH-E-1s are operating in slow speed. ☐

4.3.3 **IAAT** all of the following are < 55 psig on coolers in service.

___ RR-PI-224

___ RR-PI-225

___ RR-PI-226,

then PLACE RR-V-6 in local manual control **and THROTTLE** closed to maintain RR-PI-224, RR-PI-225 **and** RR-PI-226 > 55 psig. ☐

4.3.4 **IAAT** any of the following are > 60 psig on coolers in service

___ RR-PI-224

___ RR-PI-225

___ RR-PI-226,

then jog OPEN RR-V-5 to maintain pressure > 55 psig **and** < 60 psig. ☐ |

4.3.5 **IAAT** an individual Cooler outlet flow (Computer Points A1049, A1050, A1051) < 2200 GPM with two RR pumps or < 1600 GPM with a single RR pump, **then EVALUATE** for a potential cooler leak. ☐

4.3.6 **IAAT** a cooler leak > 50 GPM is suspected **and** at least two coolers are operable, **then PERFORM** the following: ☐

1. **OBTAIN** CRS concurrence. ___
2. **ENSURE** ESAS is BYPASSED. ___
3. **CLOSE** RR-V-3 **and** RR-V-4(s) to isolate the cooler. ___

4.3.7 **IAAT** high strainer DP alarm actuates (PPC points L2351 or L2352 alarm), **then INITIATE** Enclosure 7 of 1104-38 to manually operate strainer & backwash valves. ☐

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G2.1.27(012)	
	Importance Rating	2.8	2.9

(K&A Statement) Conduct of Operations: Knowledge of system purpose and or function.

Proposed Question: Common 13

Which ONE of the following Reactor Protection System Trips is designed to prevent exceeding Minimum DNBR limitations of the core?

A. Nuclear Overpower
B. Variable Low Pressure
C. High RCS Temperature
D. Loss of both Main FW Pumps

Proposed Answer: B. Variable Low Pressure

Explanation (Optional):

A. Plausible since coolant temperature would rise but this is designed as fuel clad protection.
B. Correct – TS Basis indicates DNB protection.
C. Plausible since high temperature could affect DNBR; however this trip is just to prevent excessive RCS temperatures.
D. Plausible since this would cause RCS temperature to rise; however no safety analysis credit is taken for this trip.

Technical Reference(s): TQ-TM-641-C001, Reactor Protection System, Page 45 (Attach if not previously provided)
TS 2.3 (Page 2-7)

Proposed references to be provided to applicants during examination: None

Learning Objective: 641-GLO-6 (As available)

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

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

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.

The high pressure trip provides protection for RCS against pressure increasing transients such as slow reactivity insertion (such as an uncontrolled slow rod withdrawal or moderator dilution) and a decrease in secondary system heat removal (such as loss or reduction of feedwater flow, or rapid reduction in steam flow).

5. Low RCS Pressure

The reactor coolant narrow range pressure signal originating from the RCS hot leg pressure transmitter is received by a buffer amplifier module in the associated protection channel, which acts as a signal conditioner.

The low pressure bistable trips when the RC pressure equals or falls below its trip setpoint (1900 psig), thus de-energizing the channel trip relay. The low RC pressure trip sets a minimum steady-state RC pressure so that transients that cause decreasing RC pressure will not violate the DNBR Safety Limit.

It is set above the high pressure injection (HPI) actuation setpoint (1600 psig) to allow normal plant shutdown.

The low pressure trip setting of 1900 psig is above the RPS shutdown bypass high pressure trip setpoint of 1720 psig by a minimum of two instrument loop error bands in order to assure a reactor trip before switching the RPS to the shutdown bypass mode.

The low pressure trip provides protection for pressure decreasing transients such as steam line breaks, steam generator tube rupture, and loss of coolant accidents (LOCA). Small break Loss of Coolant Accidents (LOCAs) have been analyzed assuming a reactor trip occurs, but large break LOCA analyses do not take credit for rod insertion to suppress reactivity. Reactivity shutdown for large break LOCAs occurs by core coolant void formation and boron injection.

This trip also assures the reactor is tripped before the Engineered Safeguards Actuation System (ESAS) is tripped on a pressure decreasing transient.

6. Variable Low Pressure

Reactor coolant outlet narrow range temperature and narrow range pressure signals are monitored by a pressure-temperature bistable.

Trip initiated when RCS pressure is less than the low pressure limit generated by the RCS Thot Narrow Range Signal Converter. Signal converter accepts a temperature input; converts it to an acceptable pressure. If RCS pressure decreases below variable calculated pressure setpoint, the bistable trips.

During the TMI-1 Cycle-7 core loading, physics data showed that the variable low pressure trip setpoint was not required for DNBR protection. However, it was reinserted in 15R to maintain margin to DNBR due to a new core design methodology with a converter setpoint of 14.29Thot -6862 psig.

This trip was originally designed to limit the minimum RC pressure as a function of RC outlet temperature. The variable setpoint of the bistable is derived to provide steady-state protection for DNBR Safety Limits. It also provides trip capability for steam generator tube rupture accidents.

7. High RCS Temperature

The Reactor Coolant (RC) outlet temperature is measured by fast responding Resistance Temperature Detectors (RTDs). The millivolt temperature signal is generated by a resistance balancing bridge. A signal converter module then converts the low level (millivolt) input signal from the balancing bridge to a 0 to 10 VDC signal that represents the narrow range hot leg temperature.

A bistable module monitors the hot leg temperature signal. When the temperature equals or exceeds the trip setpoint (618.8° F), the bistable trips de-energizing the channel trip relay.

The high reactor coolant outlet temperature trip setting limit has been established to prevent excessive core coolant temperature.

The high RC outlet temperature trip is intended to be activated by power or pressure increasing transients.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A4.01(026)	
	Importance Rating	4.5	4.3

(K&A Statement) Ability to manually operate and/or monitor in the control room: CSS controls

Proposed Question: Common 14

Plant conditions:

- Reactor tripped due to a large RCS leak (LOCA)
- ESAS actuations:
  - 4 psig RB Pressure
  - 1600 psig RCS Pressure ES signal failed to actuate.
- Current RB pressure is 20 psig
- No manual actuations were initiated
- The 4 psig ES actuation has been placed in defeat.
- No ES component positions have been changed.

Event:

- The LOCA becomes larger resulting in RB pressure rising to 35 psig.

Which ONE of the choices completes the following statement?

The Reactor Building Spray (BS) Pumps __________.

A. will start automatically, with normal flow
B. will start automatically; however the discharge valves remain closed
C. must be manually started using the BS Pump extension control switches
D. must be started by pushing the MANUAL ES 30 psig actuation pushuttons

Proposed Answer: C. must be manually started using the BS Pump extension control switches

Explanation (Optional):

A. Plausible since they would automatically start with a Block 4 ES; however, 1600 psig has failed and 4 psig is defeated.
B. Plausible if it is believed pumps start on 30 psig but 4 psig would still need to be present for valves to open.
C. Correct – no automatic signal that will start the pumps is available.
D. Plausible since this signal goes to 30 psig containment isolation valves but not the BS pumps.

Technical Reference(s): Lesson Plan TQ-TM-104-214-C001 pg 41 of 87 step 6.c

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # QR-214-GLO-10-Q03

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments:

b. The system is designed to maintain the temperature above a minimum of 60°F and below a maximum of 104°F.

5. Chemical (CA) Addition System

A nitrogen supply line connects to the gas space of the NaOH Tank. The original purpose of the line was to provide a nitrogen cover gas in the tank to prevent NaOH deterioration due to reaction with carbon dioxide contained in air. The line is currently blanked off and has not been used since initial operation of the plant.

a. The CA System provides water to the NaOH tank for making up the NaOH solution to be stored in the tank.

6. Engineered Safeguards Actuation System (ESAS)

a. Redundant Engineered Safeguards Actuation System instrument channels automatically actuate the Reactor Building Spray System on high Reactor Building pressure.

b. The Reactor Building Spray valves (BS-V-1A/B, BS-V-2A/B, BS-V3-A/B) are actuated open by the ESAS at a 4 psig. reactor building pressure setpoint.

1) BS-PT-282, 285, and 288 using a 2 out of three logic

c. The Reactor Building Spray pumps will start after the Engineered Safeguards Actuation System load sequence reaches block 4 (15 second delay) if RB pressure exceeds 30 psig. setpoint.

Refer to TMI P&IDs: SS-208-207 and SS 208-504 for controls and control logic

1) BS-PS-283, 286, and 289 using a two out of three logic will start Reactor Building Spray Pump 1A

2) BS-PS-284, 287 and 290 using a two out of three logic will start Reactor Building Spray Pump 1B.

d. Engineered Safeguards Actuation System Reactor Building pressure sensors are connected to three of the reactor building pressure sensing manifolds.

3. Reactor Protection System (RPS)

a. The Reactor Protection System trips the reactor on high reactor building pressure. The Building spray system provides a common manifold for the pressure switches that provide this function.

BS-PS-672, 673, 674 & 675

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K3.01(026)	
	Importance Rating	3.9	4.1

(K&A Statement) Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS

Proposed Question: Common 15

Which ONE of the following Reactor Building Emergency Cooler/Fan speed combinations meets accident analysis assumptions for post-accident Reactor Building emergency cooling if both Reactor Building Spray Pumps are out of service?

A. 2 Coolers with fans in SLOW speed.
B. 2 Coolers with fans in FAST speed.
C. 3 Coolers with fans in SLOW speed.
D. 3 Coolers with fans in FAST speed.

Proposed Answer: C. 3 Coolers with fans in SLOW speed.

Explanation (Optional):

A. Plausible since two is the correct combination of RB Spray Pumps if no coolers are operable and SLOW is correct.
B. Plausible since two is the correct combination of RB Spray Pumps if no coolers are operable.
C. Correct answer. TS 3.3 basis.
D. Plausible since it is the correct number but wrong fan speed.

Technical Reference(s): TS 3.3 Basis, Page 3-24 (Attach if not previously provided)
OP-TM-534-901 Page 1 step 3.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 824-GLO-7 (As available)

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

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

RB EMERGENCY COOLING OPERATIONS

1.0 PURPOSE

This procedure provides direction for emergency operations of Reactor Building Emergency Cooling System. This procedure is used for manual initiation or following automatic actuation of the system.

2.0 MATERIAL AND SPECIAL EQUIPMENT - None

3.0 PRECAUTIONS, LIMITATIONS, AND PREREQUISITES

3.1 Precautions - None

3.2 Limitations

3.2.1 To ensure containment integrity, maintain RR cooler outlet pressure (RR-PI-224, 225 & 226) between 55 and 60 psig on all in service coolers.

3.2.2 To prevent RB fan overload, operate RB fans in slow speed when RB pressure is greater than 2 psig.

3.3 Prerequisites

3.3.1 **VERIFY** Reactor Building emergency river water system was in ES standby IAW 1104-38, "Reactor Building Emergency Cooling Water System". ___

NOTE: Actuation of RBEC will discharge corrosion protection chemicals from coolers into the river. Actuation of RBEC is required to be reported to PaDER.

3.3.2 **VERIFY** 1600 psig ES actuation, RB pressure is approaching 2 psig **or** Emergency Director **or** Shift Manager has authorized use of RBEC. ___

3.3.3 **VERIFY** 1D **or** 1E 4160V Bus is energized. ___

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"

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	A1.06(039)	
Importance Rating	3.0	3.1

(K&A Statement) Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: Main steam pressure.

Proposed Question: Common 16

Plant conditions:

- Reactor at 100% power
- Turbine Bypass Valves MS-V-3A, MS-V-3B and MS-V-3C are closed in HAND for ICS module replacement

On a Reactor trip, OTSG ____ steam pressure will be _______than the normal post-trip value.

A. 1A/lower
B. 1B/lower
C. 1A/higher
D. 1B/higher

Proposed Answer: D. 1B/higher

Explanation (Optional):

A. Plausible if the examinee thinks the 1A OTSG TBVs will be open to make up for the 1B OTSG TBVs being closed and lower the 1A OTSG pressure; however the 1A OTSG TBVs will control off of Setpoint +125 psig bias.
B. Plausible if the examinee does not know MS-V-3A, B and C are off of the 1B OTSG and thinks as in 'A' above.
C. Plausible if the examinee does not know MS-V-3A, B and C are off of the 1B OTSG.
D. Correct answer. MS-V-4B will control OTSG pressure at 1026 psig.

Technical Reference(s): TQ-TM-104-621-C001, Integrated Control System (page 38) (Attach if not previously provided)

OP-TM-411-000, Main Steam/OTSG (Page 7)

Proposed references to be provided to applicants during examination: None

Learning Objective: 621-GLO-5 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

5.0 COMMITMENTS

5.1 **CM-1**, 1990T0012, Inadvertent Isolation of the Turbine Bypass Valves

5.2 **CM-2**, 1979T0014, Steam Regulating Valve Not Opened After Maintenance

5.3 **CM-3**, 1985T0054, Main Steam Safety Valve Setpoint and Performance Evaluation

5.4 **CM-4**, 1920-99-20606.003, MSIV 1B Motor Operator Failure

6.0 SYSTEM INFORMATION

6.1 Turbine Bypass and Atmosphere Dump Valve Control Options

6.1.1 Normal Operations

1. Turbine Bypass Valves (MS-V-3A-F) operated in AUTO control at Turbine Hdr Pressure setpoint plus appropriate bias (+10, +75, +125 psig) or open at 1040 whichever is the greater open signal.

2. Turbine Bypass Valves (MS-V-3A-F) operated in HAND respond only to demand changes from the ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations.

3. Atmospheric Dump Valves (MS-V-4A/B) will open at 1026 - 1052 psig.

4. Operator may select Atmospheric Dump Valves control from "MS-V-4A(B) BACKUP CTRL" stations by depressing the "B/U LOADER" pushbutton and position valves as desired.

6.1.2 Loss of Condenser Vacuum (< 2 CW pumps or < 23" Hg Vac)

1. Turbine Bypass Valves (MS-V-3A-F) are closed and latched. The ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations have no effect on their position.

2. Atmospheric Dump Valves operated in auto control at Turbine Hdr Pressure setpoint plus appropriate bias (+10, +75, +125 psig) or open at 1040 whichever demand is greater.

3. Atmospheric Dump Valves (MS-V-4A/B) operated in HAND respond only to demand changes from the ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations.

4. Operator may select Atmospheric Dump Valves control from "MS-V-4A(B) BACKUP CTRL" stations by depressing the "B/U LOADER" pushbutton and position valves as desired.

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.

Normal settings:

High Pressure Relief – 1026 PSI

Full Open at 1052 PSI

10. OTSG Press Control

The MS-V-3s can be controlled from the ICS H/A station. If in auto, they control at setpoint plus Bias or open at 1040 psig..

The MS-V-4s can be controlled from ICS H/A station (loss of condenser), Back up manual loader (loss of ICS auto power or manually selected) or Remote Shutdown Panel. The MS-V-3's Power Reset Pushbutton is located in the control room and there are select push buttons in the control room for the MS-V-4's to control from either ICS H/A station or the manual back up loaders.

a. Operation

The TBV/ADV pressure control input is from either SP6A, 6B PT1, PT2 which can be selected by the operator or by SASS.

Capacity: TBVs: 3.75% each, ADVs: 3.25% each.

ICS operation in hand is such that the TBV would will not go open without operator action. . (ADV will still open at 1026 psig if in ICS control and condenser is available).

On Loss of Condenser Vacuum the TBVs fail closed and latched. The ADV control press at header press st/pt +bias or 1040 psi if in Auto. The operator now has the capability to select the ADVs to the Manual Loader from a pushbutton in the control room. This selection to the B/U loader is possible at any time.

If TBVs are in Auto and ICS auto power is lost, control of the TBVs would switch to hand control and the ADVs automatically switch to the manual B/U loader.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A2.07(059)	
	Importance Rating	3.0	3.3

(K&A Statement) Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Tripping of MFW pump turbine.

Proposed Question: Common 17

Plant conditions:

- Reactor power 100%
- Loss of ICS Auto power occurred 10 minutes ago
- Plant stability has been verified
- Seal Injection Flow is 38 gpm
- Makeup flow is normal
- OP-TM-AOP-027, Loss of ATA or ICS Auto Power has been initiated

Subsequent event:

- Feedwater Pump FW-P-1B trips due to low oil pressure

Which ONE of the following conditions occurs and what is the required action to stabilize the plant?

A. FW-P-1B Discharge Valve will not receive an automatic close signal and must be closed manually in accordance with OP-TM-MAP-0107, "FWP 1B TRIP".

B. Flow through FW-V-17A will be excessive. Demand must be reduced in accordance with OP-TM-421-451, "Manual Control of Feed Flow to A OTSG".

C. Turbine control will be lost. A manual reactor trip is required in accordance with OP-TM-AOP-027, "Loss of ATA or ICS Auto Power".

D. Feedwater flow will be less. FW-P-1A demand must be raised to maintain the required FW Valve ΔP in accordance with OP-TM-401-472, "Manual Control of FW-P-1A".

Proposed Answer: D. Feedwater flow will be less. FW-P-1A demand must be raised to maintain the required FW Valve ΔP in accordance with OP-TM-401-472, "Manual Control of FW-P-1A".

Explanation (Optional):

A. Plausible since this action would be necessary for some FW Pump mechanical trips; however the oil pressure trip is electronic.
B. Plausible if the examinee confuses the need to reduce power due to the plants inability to run back; however valve demand will have to be raised to provide more feedwater due to the tripped pump.
C. Plausible as turbine is not controllable from ICS since trip is an action if seal injection and ICCW flow are lost; however the given conditions are that seal injection flow is 38 gpm, incorrect as turbine is controllable at OWS
D. Correct – reference directs maintaining DP at 60-90 psi.

Technical Reference(s): OP-TM-401-472, Manual Control of FW-P-1A, 3.2.3 (Page 1) (Attach if not previously provided)

OP-TM-AOP-027, Loss of ATA or ICS Auto Power (Page 3 & 5)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.5
55.43 ______

Comments:

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.1 **ANNOUNCE** entry into OP-TM-AOP-027, "Loss Of ATA Or ICS/NNI Auto Power" over the plant page and radio.	
☐ 3.2 **IAAT** Seal Injection flow **and** ICCW flow to Containment are lost, **then** perform the following: ___ 1. **PERFORM** EOP-001, "Reactor Trip" Immediate Actions. ___ 2. **TRIP** <u>all</u> Reactor Coolant pumps. ___ 3. **INITIATE** EOP-001, "Reactor Trip"	
☐ 3.3 **IAAT** Tavg < 329 °F **and** RCS pressure > 550 psig, **then LOWER** RCS pressure to < 500 psig using the PORV.	
___ 3.4 **INITIATE** OP-TM-621-471, "ICS Manual Control" to control reactor power.	
___ 3.5 **INITIATE** OP-TM-421-451 "Manual Control of Feed Flow to A OTSG".	
___ 3.6 **INITIATE** OP-TM-421-452 "Manual Control of Feed Flow to B OTSG".	
___ 3.7 **If** turbine load > 150 Mwe, **then INITIATE** OP-TM-301-471 "Manual Control of the Main Turbine" to maintain turbine header pressure using OWS LOCAL control.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.8 **INITIATE** OP-TM-401-472 "Manual Control of FW-P-1A".	
___ 3.9 **INITIATE** OP-TM-401-473 "Manual Control of FW-P-1B"	
___ 3.10 **If** MU-V-17 is in HAND, **then INITIATE** OP-TM-211-472 "Manual Pressurizer Level Control".	
___ 3.11 **INITIATE** OP-TM-211-950 "Restoration Of Letdown Flow".	
___ 3.12 **If** MU-V-32 is in HAND, **then INITIATE** OP-TM-211-476 "Seal Injection Control – MU-V-32 Console Operations".	
___ 3.13 **PLACE** pressurizer level LO LO interlock switch to BYPASS. (Key #2 inside ICS/NNI Power Monitoring Cabinet, Key #214)	
☐ 3.14 **IAAT** pressurizer level < 80" on RC-LI-777A, **then PLACE** pressurizer heaters in OFF: ___ Bank 1 ___ Bank 2 ___ Bank 3 ___ Bank 4 ___ Bank 5	

MANUAL CONTROL OF FW-P-1A

1.0 PURPOSE

1.1 Provide direction for manual control of FW-P-1A speed using ICS Hand control or Motor Speed Changer (MSC).

2.0 MATERIAL AND SPECIAL EQUIPMENT - None

3.0 PRECAUTIONS, LIMITATIONS, AND PREREQUISITES

3.1 Precautions:

3.1.1 **When** changing FW Pump speed, **then ADJUST** speed slowly enough to permit automatic or manual OTSG FW flow adjustments.

3.1.2 **When** returning both FW-P-1A and FW-P-1B to Auto, **then PLACE** FW-P-1B to Auto first.

3.2 Limitations:

3.2.1 **If** FW-P-1B is in Auto, **then MAINTAIN** FW-P-1A speed (in HAND) such that FW Pump flows are equal.

3.2.2 **If** both FW Pumps are in HAND **and** Reactor power is stable above 75%, **then MAINTAIN** Main Feedwater Valve ΔP 30 to 50 psid.

3.2.3 At all other times, **MAINTAIN** Main Feedwater Valve ΔP between 60 and 90 psid.

3.3 Prerequisites

3.3.1 **VERIFY** FW-P-1A is in Operating Mode IAW OP-TM-401-000, Main Feedwater Pumps.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K3.02(059)	
	Importance Rating	3.6	3.7

(K&A Statement) Knowledge of the effect that a loss or malfunction of the MFW will have on the following: AFW system.

Proposed Question: Common 18

Initial plant conditions:

- Reactor power 100%
- Condensate/Booster Pump String "C" (CO-P-1C and CO-P-2C) OOS for maintenance
- All other equipment in a normal lineup

Event:

- 1A Aux Transformer fault with failure of the 1A and 1B 4160 Volt Bus automatic transfers
- Feedwater Pump 1A failed to trip and the CRO reduced speed to zero

Which ONE of the following identifies the response of the Emergency Feedwater System?

A. Only EF-P-1 and EF-P-2B start automatically. Level must be controlled manually in OTSG "A".
B. Only EF-P-1 and EF-P-2B start automatically. Level will be controlled automatically in both OTSGs.
C. All Emergency Feedwater Pumps start automatically. Level must be controlled manually in OTSG "A".
D. All Emergency Feedwater Pumps start automatically. Level will be controlled automatically in both OTSGs.

Proposed Answer: D. All Emergency Feedwater Pumps start automatically. Level will be controlled automatically in both OTSGs.

Explanation (Optional):

A. Plausible if the examinee does not recognize all Emergency Feedwater Pumps will start on low OTSG level in both OTSGs.
B. Plausible if the examinee does not recognize all Emergency Feedwater Pumps will start on low OTSG level in both OTSGs. The second part is correct.
C. Plausible if the examinee thinks the failure of FW-P-1A trip circuit will prevent automatic control in OTSG "A".
D. Correct answer. All EFW pumps and both HSPS trains will actuate at 10" in the OTSGs. Level will be controlled automatically at 25".

Technical Reference(s):	MAP Alarm J-2-3, OTSG A Level LO	(Attach if not previously provided)
	MAP Alarm J-2-4, OTSG B Level LO	

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

J-2-3
Revision 17

ALARM:

OTSG A LEVEL LO

SET POINTS:

1. LAL-1042/HSPS two of four HSPS S/U Lvl CH < 20"

CAUSES:

1. Improper operation of FW-V-16A or 17A in "Auto" or "Hand"
2. FW-V-5A or 92A not open when req'd
3. Inadequate FW pump 1A and 1B discharge pressure
4. Loss of FW system or OTSG integrity

AUTOMATIC ACTION:

1. ICS will attempt to maintain OTSG level at ≥ Low Level Limits (25")
2. If S/U Lvl is < 10" on 2 of 4 HSPS Channels EF-P-2A, 2B and EF-P-1 will start. EF-V-30A and D will open to control Lvl at 25".

OBSERVATION (CONTROL ROOM):

1. Start up level on recorder on CC and digital on CL
2. FW-V-16A and 17A demand and FW-V-5A and 92A position
3. Start up flow indication on CC
4. "A" FW valve ΔP on CL
5. HSPS status lights (lit if channel is < 10") on CC
6. RCS pressure on CC and PLF

MANUAL ACTION REQUIRED:

CAUTION

If taking manual action with FW-V-16A/17A, do so carefully coordinating with RCS parameters so as not to cause a Reactor trip.

1. Verify proper operation of FW-V-16A and 17A if necessary take "Hand" control and maintain desired level at greater than LLL.
2. Verify proper operation of FW-P-1A/1B.
3. Consult OP-TM-EOP-010 for guidance for feeding dry OTSG. (i.e. "A" OTSG press is 200 psig < SAT press for Loop A T_{cold}) & ≤ 0 inches on S. U. range.

J-2-4
Revision 17

ALARM:

OTSG B LEVEL LO

SET POINTS:

1. LAL-1054/HSPS two of four HSPS S/U Lvl CH < 20"

CAUSES:

1. Improper operation of FW-V-16B or 17B in "Auto" or "Hand"
2. FW-V-5B or 92B not open when req'd
3. Inadequate FW pump 1A and 1B discharge pressure
4. Loss of FW system or OTSG integrity

AUTOMATIC ACTION:

1. ICS will attempt to maintain OTSG level at ≥ Low Level Limits (25")
2. If S/U Lvl is < 10" on 2 of 4 HSPS Channels EF-P-2A, 2B and EF-P-1 will start. EF-V-30B and C will open to control Lvl at 25".

OBSERVATION (CONTROL ROOM):

1. Start up level on recorder on CC and digital on CL
2. FW-V-17B and 16B demand and FW-V-5B and 92B position
3. Start up flow indication on CC
4. "B" FW valve ΔP on CL
5. HSPS status lights (lit if channel is < 10") on CC
6. RCS pressure on CC and PLF

MANUAL ACTION REQUIRED:

CAUTION

If taking manual action with FW-V-16B/17B, do so carefully coordinating with RCS parameters so as not to cause a Reactor trip.

1. Verify proper operation of FW-V-16B and 17B if necessary take "Hand" control and maintain desired level at greater than LLL.
2. Verify proper operation of FW-P-1A/1B.
3. Consult OP-TM-EOP-010 for guidance for feeding dry OTSG. (i.e., "B" OTSG press is 200 psig < SAT press for Loop B T_{cold}) & ≤ 0 inches on S. U. range.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K6.02 (061)	
	Importance Rating	2.6	2.7

(K&A Statement) Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps.

Proposed Question: Common 19

Plant conditions:

- Emergency Feedwater Pump EF-P-1 OOS
- OTSG B Tube Rupture occurs while at 100% power
- Reactor tripped
- Subcooling Margin was lost and OP-TM-EOP-010 Rule 1, "Loss of Subcooling Margin (SCM)" was initiated
- Loss of offsite power (LOOP) with a failure of the EG-Y-1B Emergency Diesel to start
- Emergency Feedwater Pump EF-P-2A is feeding both OTSGs as follows:
  - OTSG A - 312 gpm
  - OTSG B - 210 gpm

In Accordance with OP-TM-EOP-010 Rule 4, "FWC Feedwater Control", which ONE of the following actions is required?

A. Stop feeding OTSG "B" and throttle Emergency Feedwater Control Valve EF-V-30A to feed "A" OTSG at between 430 gpm and 515 gpm.
B. Stop feeding OTSG "B" and fully open Emergency Feedwater Control Valve EF-V-30A to feed OTSG "A" at the maximum available rate.
C. Throttle Emergency Feedwater Control Valves EF-V-30A and EF-V-30B to feed each OTSG at >215 gpm each.
D. Throttle Emergency Feedwater Control Valves EF-V-30A and EF-V-30B such that total emergency feedwater flow to both OTSG's is greater than 430 gpm.

Proposed Answer: A. Stop feeding OTSG "B" and throttle Emergency Feedwater Control Valve EF-V-30A to feed "A" OTSG at between 430 gpm and 515 gpm.

Explanation (Optional):

A. Correct – feed is isolated to the ruptured OTSG and maintained at ≥ 430 for heat removal but ≤515 gpm for runout.
B. Plausible since the first part is correct but runout criteria of Guide 15 is neglected.
C. Plausible since this is the value for SCM < 25 °F but OTSG leakage < 1 gpm.
D. Plausible since this is the correct value for the good OTSG.

Technical Reference(s):	OP-TM-EOP-010, Rule 4, Feedwater Control (Page 8)	(Attach if not previously provided)
	OP-TM-EOP-010, Guide 15, EFW Actuation Response (Page 26)	

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments:

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. **VERIFY** SCM > 25°F.	**If** both OTSGs are available **and** OTSG tube leak < 1 gpm, **then FEED** with EFW > 215 gpm / OTSG. **If** only one OTSG is available **or** OTSG tube leak > 1 gpm, **then FEED** with EFW > 430 gpm to the good OTSG.
2. **VERIFY** an RCP is operating **or** incore temperature is stable or lowering.	**FEED** OTSG at maximum available EFW flow.
3. **VERIFY** EFW is available.	**If** SCM < 25°F, **or** [all RCPs OFF **and** incore temp is rising], **then FEED** with MFW at > 1.0 Mlbm/hr.
4. There is no minimum required flow rate.	

Guide 15
EFW Actuation Response

IAAT EFW is actuation is required, **then**:

____ 1. **ENSURE** EF-P-1, EF-P-2A, **and** EF-P-2B start.

____ 2. **DISPATCH** an Auxiliary Operator (AO) to EF-V-30 area.

____ 3. **If** EFW pump disch. pressure < OTSG pressure, **then INITIATE** Guide 16.

☐ 4. **IAAT** OTSG pressure is more than 100 psig below desired, **then THROTTLE** EFW flow if permitted by Rule 4

☐ 5. **IAAT** only EF-P-2A **or** EF-P-2B is operating, **then THROTTLE** EF-V-30s to maintain total EFW flow < 515 GPM.

____ 6. **If** EFW is being manually initiated **and** the OTSG is DRY, **then VERIFY** Guide 13, DRY OTSG requirements are satisfied.

____ 7. **ENSURE** EF-V-30A/D and EF-V-30B/C control OTSG level at setpoint (Rule 4).

____ 8. **If** Shift Management concurrence is obtained, **then** EFW flow may be controlled using one **or** both EF-V-30 valves for each OTSG.

☐ 9. **IAAT** an EFW failure occurs, **then INITIATE** Guide 16.

☐ 10. **IAAT** EF-P-1 operation is required,

  ____ 1 **If** OTSG Tube leakage symptoms exist **and** aux steam is available, **then OPEN** AS-V-4.

  ____ 2 **If** OTSG A **and** B pressure < 200 psig, **then**,

    ____ **CLOSE** the breaker for MS-V-10A [1C DC switch #1]

    ____ **CLOSE** the breaker for MS-V-10B [1D DC switch #1]

    ____ **Jog OPEN** MS-V-10A **or** B to maintain EF-P-1 speed > 3300 RPM.

  ____ 3 **If** OTSG A **and** B pressure < 150 psig **and** aux steam is available, **then OPEN** AS-V-4.

____ 11. **ENSURE** AH-E-24A **or** AH-E-24B is operating.

____ 12. **When** CO-T-1A **or** CO-T-1B < 5 ft, **then INITIATE** Guide 17.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	A3.01(062)	
Importance Rating	3.0	3.1

(K&A Statement) Ability to monitor automatic operation of the ac distribution system, including: Vital ac bus amperage

Proposed Question: Common 20

Maintenance requires 1P 480 volt bus to be fed from 1S 480 volt bus.

In accordance with OP-TM-731-550, "Cross Tie of ES 480 Volt Bus Feeding 1P From 1S", current to the 1P 480 volt bus is monitored where, and by what means?

A. 4160 V Line to 1P Bus, by calculating.
B. 480 V Line to 1P Bus, by direct reading.
C. 4160 V Line to 1S Bus, by calculating.
D. 480 V Line to 1S Bus, by direct reading.

Proposed Answer: C. 4160 V Line to 1S Bus, by calculating.

Explanation (Optional):

A. Plausible because this is the normal measuring point for 1P.
B. Plausible since this would be correct normal path if amperage indication was available.
C. Correct – current cannot be monitored directly and is determined from the 4160 V feeding bus.
D. Plausible since the limit applies to 1S but no ammeter exists.

Technical Reference(s): OP-TM-731-550, Cross Tie of ES 480 Volt Bus Feeding 1P From 1S (Pages 2) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ____

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments:

3.3 Prerequisites

3.3.1 Additional compensatory measures required to support bus outage have been listed on Attachment 7.2. ____

3.3.2 Refer to Technical specifications to determine any actions that may be required, due to the loss of electrical separation, which negates the required redundancy caused by the cross tie of the 480 volt ES busses. Verify reactor is shutdown. ____ |

3.3.3 Verify that the load to be picked up is less than 640 amps, or is within the capability of the Emergency Diesel, if it is the sole provider of power to the supplying 4KV bus. ____

NOTE

640 amps of load at 480 volts corresponds to 74 amps of load as read on the 4KV bus. Since 480 volt bus amps are not metered in the control room, monitor the amps read on the ammeter for the supplying breaker on Console Right.

3.3.4 Verify that there is DC control power available for the breakers to be operated by observing the lights above the control switches in the Control Room. ____

3.3.5 Operations Shift Management has:

- Reviewed Attachment 7.1 and Bus Casualty / Electrical Distribution Panel Listings in 1107-4 to understand the scope of equipment that will be de-energized and cross tied. ____
- Specified any applicable Technical Specification limitations or other limitations in affect due to current plant conditions. ____
- Listed on Attachment 7.2 any additional compensatory measures to be implemented due to current plant conditions. ____
- Granted permission to tie 1P 480 volt bus to 1S 480 volt bus. ____

______________________________ ____________/__________

Operations Shift Management Date / Time

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	A3.01(063)	
Importance Rating	2.7	3.1

(K&A Statement) Ability to monitor automatic operation of the dc electrical system, including: Meters, annunciators, dials, recorders, and indicating lights.

Proposed Question: Common 21

For the 1A Station Battery to be considered operable....

A. Battery Charger 1E must be operable.
B. equalizing charge current must be < 2 amps.
C. Battery Room ventilation must be in operation while the reactor is critical.
D. terminal voltage must be between 129 and 131 VDC when not on an equalizing charge.

Proposed Answer: D. terminal voltage must be between 129 and 131 VDC when not on an equalizing charge.

Explanation (Optional):

A. Plausible since two Battery Chargers must be operable; however the spare does not have to be one of the two.
B. Plausible since the float charge must be <2 amps for the battery to be considered operable.
C. Plausible since the ventilation system must be in operation during a battery equalizing charge or must be stopped within 2 hours if the ventilation system is out of service; however it does not have to be in service to consider the battery operable.
D. Correct – per procedure Limits and Precautions.

Technical Reference(s): OP 1107-2C, Vital DC Electrical, 2.0.I (Pages 3 & 4) (Attach if not previously provided)

TS 3.7.1.f, Unit Electric Power System

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ________

Comments:

1.0 **REFERENCES**

1.1 Systems Description

1.1.1 Emergency Electrical System – Volume 1

1.1.2 FSAR – Three Mile Island, Unit 1

1.2 Operating Procedures

1.2.1 1107-1 Normal Electrical System

1.2.2 1107-2A Emergency Electrical – 4KV and 480 volt

1.2.3 1107-2B 120 volt Vital Electrical System

1.3 Technical Specifications

1.3.1 Section 3.7 Electric Power System

1.3.2 Section 4.6 Emergency Power System Periodic Tests

1.4 Vendor Manuals

1.4.1 VM-TM-2646, Station Battery Ground Fault Detector

2.0 **LIMITS AND PRECAUTIONS**

A. Adhere to safety rules and regulations for operation and maintenance of electrical equipment.

B. Maintenance must be scheduled so as not to conflict with the requirements of Technical Specifications.

C. Battery room Ventilation must be maintained during equalizing battery charging operations. If Battery room ventilation is lost during an equalizing charge, within 2 hours, secure the equalizing battery charge and do not recommence until ventilation is restored.

D. Anytime HPI must be operable, with the HPI Discharge cross connects closed between MU-P-1B and MU-P-1C (MU-V-76A/B are closed), the power source to the 1M DC panel should be selected to the "A" DC system. If the HPI discharge cross connects are closed between MU-P-1A and MU-P-1B (MU-V-77A/B are closed), then the 1M DC panel should be powered from the 1B DC system.

E. When starting up a battery charger, it is possible to damage the rectifier stack or blow the anode fuse if the DC breaker is not closed first.

F. When returning a battery charger to service, the battery chargers will be in parallel operation for a short time. A voltage spike during parallel operation could cause a high voltage trip on one or both battery chargers.

G. Equalizing Battery charges should not be continued for greater then 24 hours without approval of the Electrical Maintenance Foreman.

H. DO NOT CLOSE THE BATTERY CROSS TIE WHILE THE REACTOR IS CRITICAL.

I. A station battery is OPERABLE when it is (1) connected to the associated DC distribution train, (2) the battery terminal voltage (as read at battery ground detectors) is between 129.0 and 131.0 VDC (or 134 or 136 VDC if on equalizing charge), (3) the individual cell voltages are greater than 2.07 VDC and (4) the float charging current is less than 2 amps.

3.0 <u>OPERATING PROCEDURES</u>

> **NOTE**
>
> The DC Vital system is normally energized. The operating procedures in this section provide the necessary guidance for approved deviation from the normal electrical lineup and returning it to a normal configuration.

3.1 **Removing a Battery Charger from service - Level 1**

3.1.1 Prerequisites

_____ 1. IF at power, THEN verify that there are at least two battery chargers in service per side. If there are not, then do not proceed with this procedure without SM direction.

> **NOTE**
>
> The 1E battery charger is capable of supplying either 125VDC panel on the "A" DC System. The 1F battery charger is capable of supplying either 125VDC panel on the "B" DC System.

_____ 2. Verify that each charger is supplying a separate 125VDC panel on the applicable 250VDC panel.

3.7 UNIT ELECTRIC POWER SYSTEM

Applicability
Applies to the availability of electrical power for operation of the unit auxiliaries.

Objective
To define those conditions of electrical power availability necessary to ensure:

a. Safe unit operation

b. Continuous availability of engineered safeguards

Specification

3.7.1 The reactor shall not be made critical unless all of the following requirements are satisfied:

a. All engineered safeguards buses, engineered safeguards switchgear, and engineered safeguards load shedding systems are operable.

b. One 7200 volt bus is energized.

c. Two 230 kV lines are in service.

d. One 230 kV bus is in service.

e. Engineered safeguards diesel generators are operable and at least 25,000 gallons of fuel oil are available in the storage tank.

f. Station batteries are charged and in service. Two battery chargers per battery are in service.

3.7.2 The reactor shall not remain critical unless all of the following requirements are satisfied:

a. Offsite Sources:

(i.) Two 230 kV lines are in service to provide auxiliary power to Unit 1, except as specified in Specification 3.7.2e below.

(ii.) The voltage on the 230 kV grid is sufficient to power the safety related ES loads, except as specified in Specification 3.7.2.h below.

b. Both 230/4.16 kV unit auxiliary transformers shall be in operation except that within a period not to exceed eight hours in duration from and after the time one Unit 1 auxiliary transformer is made or found inoperable, two diesel generators shall be operable, and one of the operable diesel generator will be started and run continuously until both unit auxiliary transformers are in operation. This mode of operation may continue for a period not exceeding 30 days.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A1.03(064)	
	Importance Rating	3.2	3.3

(K&A Statement) Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ED/G system controls including: Operating voltages, currents, and temperatures.

Proposed Question: Common 22

Plant conditions:

- Reactor power 100%
- Emergency Diesel Generator EG-Y-1A monthly surveillance run is in progress
- EG-Y-1A voltage control is in automatic
- MVAR reading on the diesel is zero
- Indicated 1D 4160 Volt ES Bus voltage is 4150 volts
- EG-Y-1A manual voltage setting is at 4165 volts

Transfer of EG-Y-1A Voltage Control to MANUAL in this situation will result in …

A. bus voltage rising.
B. bus voltage lowering.
C. reactive load on the diesel rising.
D. reactive load on the diesel lowering.

Proposed Answer: C. reactive load on the diesel rising.

Explanation (Optional):

A. Plausible if the examinee thinks the diesel can affect bus voltage while loaded in parallel with the grid.
B. Plausible if the examinee thinks the diesel can affect bus voltage while loaded on the grid.
C. Correct – reactive load will increase with voltage when the diesel is synchronized.
D. Plausible if the examinee does not understand the relationship between EDG voltage and the grid.

Technical Reference(s): OP 1107-3, Diesel Generator (Page 27) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ________

Comments:

____ 11. Adjust diesel engine speed, using the EG-Y-1A(1B) GOVERNOR control switch, until synchroscope is rotating slowly (less than one revolution per 15 seconds) in the fast (clockwise) direction.

CAUTION

If diesel generator voltage or frequency is less than bus voltage when the breaker is closed, the diesel generator may trip on reverse power.

12. With the EXCITER switch in **MANUAL**, or placed in **AUTO** if desired, perform one of the following:

____ a. If controlling voltage manually: Adjust **MANUAL** VOLTAGE CONTROLLER on console CR until the generator voltage matches or exceeds bus voltage (-0 to +.05 KV) on synchronizing voltmeter (otherwise N/A).

____ b. If controlling voltage in automatic: Request the auxiliary operator to adjust the VOLT CONT RHEOSTAT on the outside of the EG-Y-1A(1B) Local Alarm Panel until the generator voltage matches or exceeds bus voltage (-0 to +.05 KV) on the synchronizing voltmeter (otherwise N/A).

CAUTION

If system voltage and/or frequency stability is uncertain, do not parallel the D/G to the system without concurrence by the Director, Operations.

____ 13. Verify that no problems exist with the engine, as reported by the auxiliary operator.

____ 14. Verify that voltage and synchroscope rotation are acceptable and that the READY-TO-LOAD light is on.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A2.02(073)	
	Importance Rating	2.7	3.2

(K&A Statement) Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question: Common 23

Plant conditions:

- Plant is in cold shutdown preparing to go to refueling shutdown
- Reactor Building purge is in progress
- Waste Gas Tank 1C is being released
- The Reactor Building Sump is being drained for inspection

Event:

- C-1-1, "RM-A-9 HI (RB STACK MONITOR)" high alarm is received
- RM-A-9 P,I,G channels are rising slowly but below alert setpoints

Which ONE of the following actions is required by MAP C-1-1?

A. Close the RB Purge Valves.
B. Close Waste Gas Stop Valve WDG-V-47.
C. Verify Control Tower on Pressurized Recirculation.
D. Verify the RB Purge Supply and Exhaust fans tripped automatically.

Proposed Answer: A. Close the RB Purge Valves.

Explanation (Optional):

A. Correct answer per MAP C-1-1, Page 17.
B. Plausible if the examinee thinks the gas release should be terminated if hi activity exists in the RB or Auxiliary Building exhaust.
C. Plausible since this would be a correct action for an RM-A-1, and would limit the possibility of airborne contamination entering the control room.
D. Plausible since securing the RB Purge fans would be an action to take if the Purge Valves were to close; however there are no interlocks associated with RM-A-9 HI alarm.

Technical Reference(s): C-1-1, RM-A-9 HI (RB STACK MONITOR), Page 17 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.10

55.43 ______

Comments:

ALARM:

RM-A-9 HI (RB STACK MONITOR)

SET POINTS:

Refer to Operating Procedure 1101-2.1 RMS setpoints.

CAUSES:

Purge valves fail to close on RM-A-9 gas "Hi" Alarm.

AUTOMATIC ACTION:

None

OBSERVATION (CONTROL ROOM):

1. RM-A-9 Hi "Alert" Alarm on PRF.
2. RM-A-9 Hi "Hi" Alarm on PRF.
3. RM-A-9 Hi Indication on PRF > above setpoints.

MANUAL ACTION REQUIRED:

1. Close AH-V-1A, 1B, 1C, 1D, WDL-V-534, 535 if not already closed.
2. Refer to EP 1202-12, Excessive Radiation Levels.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K1.05 (006)	
	Importance Rating	2.8	2.9

(K&A Statement) Knowledge of the physical connections and/or cause-effect relationships between the ECCS system and the following systems: RCP seal injection and return

Proposed Question: Common 24

Plant conditions:

- A large break LOCA resulted in a reactor trip and ESAS actuation.
- RCS and Reactor Building pressure have equalized at 40 psig.

Which ONE of the choices completes the following statement?

The Reactor Coolant Pump seal injection flowpath is __________ and the seal return flowpath is __________.

A. open; open
B. open; isolated
C. isolated; open
D. isolated; isolated

Proposed Answer: B. open; isolated

Explanation (Optional):
A. Plausible because the first part is correct.
B. Correct. On a 30 psig signal, the return is isolated. Injection stays in service.
C. Plausible because it is the opposite.
D. Plausible because the second half is correct.

Technical Reference(s): OP-TM-244-901, Page 13 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: __________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

ATTACHMENT 7.1
Containment Isolation Verification
Page 2 of 2

7.3 **VERIFY** the following 1600 psig / ICCW surge tank level interlock Containment isolation valves are closed:

	✓		✓		✓		✓
IC-V 3		**IC-V 4**		**IC-V 6**		**IC-V 2**	

7.4 **VERIFY** the following 1600 psig / NSCW surge tank level interlock Containment isolation valves are closed:

	✓		✓		✓
NS-V 4		**NS-V 15**		**NS-V 35**	

7.5 **VERIFY** the following 30 psig Containment isolation valves are closed:

	✓		✓		✓
MU-V 25		**NS-V 4**		**IC-V 3**	
	✓		✓		✓
NS-V 15		**IC-V 4**		**IC-V 6**	
	✓		✓		✓
MU-V 25		**NS-V 35**		**IC-V 2**	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G2.1.2(076)	
	Importance Rating	3.0	4.0

(K&A Statement) Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: Common 25

Which ONE of the following Secondary Closed Cooling Water (SCCW) System alignments requires permission from the Director of Operations?

A. One SCCW Pump running with two SCCW Coolers in service.
B. Three SCCW Pumps running with four SCCW Coolers in service.
C. Two SCCW Coolers with closed side flow throttled utilizing closed side outlet valves.
D. Two SCCW Coolers in service with one having the river water outlet valve closed and river water inlet valve open.

Proposed Answer: B. Three SCCW Pumps running with four SCCW Coolers in service.

Explanation (Optional):

A. Plausible because this is a permissible alignment per the same L&P as the one requiring special permission.
B. Correct – pump capacity exceeds cooler capability but can be used with permission when river water temperature is excessive.
C. Plausible because this is a permissible alignment per the same L&P as the one requiring special permission.
D. Plausible because this is a permissible alignment per the same L&P as the one requiring special permission.

Technical Reference(s): OP 1104-12, Secondary Services Closed Cooling Water Systems L&P 2.1.3, Bullet 4 (Page 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 b.10
55.43

Comments:

NRC QU comments

		Number
	TMI - Unit 1 Operating Procedure	**1104-12**
Title		Revision No.
Secondary Services Closed Cooling Water Systems		**53**

2.0 LIMITS AND PRECAUTIONS

2.1 Equipment

2.1.1 Do not operate the Secondary Service Closed Cooling Water Pumps for extended periods of time with the discharge valve closed.

2.1.2 Monitor all rotating machinery and component equipment coolers on a routine basis.

- SCCW Pump oil reservoir level should be maintained at the high mark on the sight glass when the pump is not operating and at the medium mark when in operation.

2.1.3 In order to balance heat load and heat rejection and to prevent excessive Secondary Services Cooler tube vibration and attendant tube failures the following must be observed:

- Prior to establishing secondary closed flow through a heat exchanger, ensure that the river water side inlet valves are open.
- Secondary closed side cooler inlet valves should be kept open on all available coolers.
- If it becomes necessary to throttle closed side flow, use the closed side outlet valve.
- Do not put more than one pumps worth of flow through a heat exchanger. By having one more cooler in operation than the number of pumps operating this requirement can be met and still allow throttling on the extra cooler. The acceptable pump/cooler combinations are 1 pump/2 coolers, 2 pumps/3 coolers. 3 pumps/4 coolers is not a designed combination, since each pump is designed for 50% capacity and each cooler is designed for 33 1/3% capacity. If 3 pumps/4 coolers operation is desired due to excessively high river water temperature (greater than 91 degrees F), then obtain permission from the Director of Operations. 3 pumps/4 coolers operation is to be minimized. During 3 pumps/4 coolers operations heat exchanger tube rattle is to be monitored. If tube rattle occurs, return to 2 pumps/3 coolers operations.
- During cold weather operation when secondary services closed flow is throttled to the point that a pumps worth of flow is passing through a heat exchanger (equivalent number of coolers with valves full open equals number of operating pumps) it is permissible to throttle the river water outlet valve. It is also permissible to have the river water outlet valve closed with closed side flow as long as the river water inlet valve is open and there is no excessive river water flow on the other Secondary Services coolers. See OP 1104-31 (Secondary River Water).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K1.04 (078)	
	Importance Rating	2.6	2.9

(K&A Statement) Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor.

Proposed Question: Common 26

Which ONE of the following describes the condition when Fire Service Water aligns to cool Instrument Air Compressors?

A. Low SCCW System pressure.
B. High SCCW System temperature.
C. Tripping of all 3 SCCW Pump breakers.
D. High Instrument Air Compressor temperature.

Proposed Answer: C. Tripping of all 3 SCCW Pump breakers.

Explanation (Optional):

A. Plausible since low pressure is often a symptom associated with low flow.
B. Plausible since high system temperature is often a symptom associated with low flow.
C. Correct – SC-V-57A/B and SC-V-58A/B open on interlock (breaker contacts).
D. Plausible since high system temperature is often a symptom associated with low flow.

Technical Reference(s): TQ-TM-104-850, Station Air Systems (Page 9) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

b) IA-V-2714 (check valve)

c) SA-V-232 (manual isolation valve)

e. Standby Instrument Air Compressors IA-P-1A/B

Refer to flow diagram 302-271

1) Located in separate cubicles in the Intermediate Building Basement

2) Two 100% capacity compressors

a) Approximately 250 scfm per compressor

b) Non-Lubricated single piston

(1) Teflon Piston Rings

(2) Graphite Wear Runners

(3) V belt driven

c) Normally cooled by Secondary Services Closed Cooling Water (SSCCW)

(1) Never operate unless cooling water is established

(2) Fire Service Water provides backup cooling in an emergency, if loss of all 3 Secondary Services Closed Cooling Water Pumps, sensed from breaker contacts

d) Do NOT add oil to the motor when the compressor is running

f. Intake Filters/Silencers IA-F-1A and IA-F-18

1) Located in Intermediate Building basement on the suction of the associated Standby Instrument Air Compressor

2) Dry wool felt element

g. After-coolers IA-C-1A and IA-C-18

1) Located in Intermediate Building basement in associated Standby Instrument Air Compressor cubicle, east wall

2) Removable tubes

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G2.1.33(103)	
	Importance Rating	3.4	4.0

(K&A Statement) Conduct of Operations: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

Proposed Question: Common 27

Plant conditions:

- 100% power

Which ONE of the following requires action in accordance with Technical Specifications?

A. Reactor Building pressure is 0.5 psig vacuum.
B. Reactor Building Ventilation Fan AH-E-1C is tripped.
C. Reactor Building Purge Isolation AH-V-1A fails to close when a Reactor Building purge is secured.
D. Reactor Building Emergency Cooling System Cooler 1A has been secured and isolated to stop a cooling water a leak.

Proposed Answer: C. Reactor Building Purge Isolation AH-V-1A fails to close when a Reactor Building purge is secured.

Explanation (Optional):

A. Plausible if a candidate does not know that we are permitted to operate at a vacuum.
B. Plausible since two fans could be out of service at one time with the second being out for seven days; however one fan can be out of service indefinitely.
C. Correct – containment isolation valve specifically identified in TS 3.6.8.
D. As in B above two coolers could be out of service at one time with the second being out for seven days; however one cooler can be out of service indefinitely

Technical Reference(s): TS 3.6.8 (Page 3-41a) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments:

3.6 <u>REACTOR BUILDING (Continued)</u>

3.6.7 DELETED

3.6.8 While CONTAINMENT INTEGRITY is required (see Specification 3.6.1), if a 48" reactor building purge valve is found to be inoperable perform either 3.6.8.1 or 3.6.8.2 below.

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage, close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared OPERABLE. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (see Specification 6.8.5), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below.

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of Specification 6.8.5 and perform either (1) or (2) below:

(1) Restore the leaking valve to OPERABILITY within the following 72 hours.

(2) Maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test in accordance with the Reactor Building Leakage Rate Testing Program. In order to accomplish repairs, one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V-1A&D) shall be limited to less than 31 degrees and (AH-V-1B&C) shall be limited to less than 33 degrees open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons.

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A3.01(103)	
	Importance Rating	3.9	4.2

(K&A Statement) Ability to monitor automatic operation of the containment system, including: Containment isolation.

Proposed Question: Common 28

Plant conditions:

- Reactor tripped due to a loss of offsite power (LOOP)
- Emergency Diesel Generator 1B did not start on the LOOP
- A subsequent large RCS leak (LOCA) developed in the Reactor Building
- Reactor Building pressure peaked at 34 psig
- ESAS signals actuated as expected for the plant conditions
- NS-V-15 Nuclear Services Supply Valve is verified closed
- IC-V-4 to RB components Supply Valve is verified closed
- IC-V-6 to CRD cooling Supply Valve is verified closed

With respect to Intermediate Closed Cooling and Nuclear Services Closed Cooling, to satisfy containment integrity, ensure _________ are closed?

A. IC-V-2, ICCW From RB, and NS-V-35, RCP Motor Cooler Outlet Isolation.
B. IC-V-3, ICCW From RB, and NS-V-4, NS from RCP motor.
C. IC-V-2, ICCW From RB, and NS-V-4, NS from RCP motor.
D. IC-V-3, ICCW From RB, and NS-V-35, RCP Motor Cooler Outlet Isolation.

Proposed Answer: B. IC-V-3, ICCW From RB, and NS-V-4, NS from RCP motor.

Explanation (Optional):

A. Plausible as these are containment valves however they are powered from 1B ESV and have no power.
B. Correct – both are required to be closed with IC-V-2 and NS-V-35 open
C. Plausible second part correct first part a reversal.
D. Plausible first part correct second part a reversal.

Technical Reference(s): OP-TM-244-901, Containment Isolation, 4.2.15 & 16 (Page 5,6) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	______	(As available)	
Question Source:	Bank #	______	
	Modified Bank #	______	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge	______	
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.9	
	55.43	______	

4.2.12 **If** CA-V-189 is **not** Closed, **then** perform the following: ___

1. **ENSURE** CA-V-189 is Closed **or**, ___

2. **CLOSE** IA-V-840 (AB 281: RB wall near BS-V-60B) to fail CA-V-189 Closed. ___

4.2.13 **If** 1600 psig or 4 psig ESAS was required, **then** perform the following: ___

1. **ENSURE** RB-V-2A is Closed. ___

2. **ENSURE** RB-V-7 is Closed. ___

3. **If** CM-V-1 or CM-V-2 is **not** Closed, **then** perform the following: ___

   A. **ENSURE** CM-V-1 **or** CM-V-2 is Closed **or**, ___

   B. **CLOSE** IA-V-2800 **or** IA-V-2801 to fail CM-V-1 or CM-V-2 Closed. (IB 295: Room south of RM-A-2, 2 ft from ceiling) ___

4. **If** CM-V-3 or CM-V-4 is **not** Closed, **then** perform the following: ___

   A. **ENSURE** CM-V-3 **or** CM-V-4 is Closed **or**, ___

   B. **CLOSE** IA-V-2802 **or** IA-V-2803 to fail CM-V-3 or CM-V-4 Closed. (IB 295: Room south of RM-A-2, 2 ft from ceiling) ___

4.2.14 **If** 30 psig ESAS was required, **then** perform the following: ___

1. **If** MU-V-25 or MU-V-26 is **not** Closed, **then** perform the following: ___

   A. **ENSURE** MU-V-25 **or** MU-V-26 is Closed **or**, ___

   B. **CLOSE** IA-V-1214 to fail MU-V-26 Closed. (AB 305: Near fuel transfer tubes) ___

4.2.15 **If** 30 psig ESAS or NSCCW line break was required, **then** perform the following: ___

1. **ENSURE** NS-V-15 is Closed ___

2. **If** NS-V-4 **or** NS-V-35 is **not** Closed, **then ENSURE** NS-V-4 **or** NS-V-35 is Closed. ___

4.2.16 **If** 30 psig ESAS or ICCW line break was required, **then** perform the following: ___

1. **ENSURE** IC-V-4 is Closed. ___
2. **ENSURE** IC-V-6 is Closed. ___
3. **If** IC-V-2 **or** IC-V-3 is **not** Closed, **then** perform the following: ___
   A. **ENSURE** IC-V-2 **or** IC-V-3 is Closed or. ___
   B. **CLOSE** IA-V-843 (AB 281: RB wall under IC-V-3) to fail IC-V-3 Closed. ___

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	K5.33 (001)	
	Importance Rating	3.2	3.5

(K&A Statement) Knowledge of the following operational implications as they apply to the CRDS: Xenon production and removal process

Proposed Question: Common 29

Plant conditions:

- Reactor Power 100% for 200 days.

Event:

- One Control Rod in Group 6 drops to the fully inserted position.
- ICS runback operates as expected to reduce power to approximately 455 MWe.
- Initial attempts to recover the rod have failed.

To maintain power level at 55 % over the next 24 hours, in order to compensate for Xenon it will be necessary to perform a combination of either ...(both choices must be correct)

A. insert rods and then withdraw rods OR add boron and then reduce boron concentration.
B. withdraw rods and then insert rods OR reduce boron concentration and then add boron.
C. insert rods and then withdraw rods OR reduce boron concentration and then add boron.
D. withdraw rods and then insert rods OR add boron and then reduce boron concentration.

Proposed Answer: B. withdraw rods and then insert rods OR reduce boron concentration and then add boron.

Explanation (Optional):

A. Plausible if production and removal of Xenon is misunderstood. Incorrect as Xenon builds in rods must be withdrawn or boron reduced.
B. Correct during the first few hours Xenon builds in, then decays off to a new lower equilibrium value.
C. Plausible correct boration choice, wrong rod choice.
D. Plausible correct rod choice wrong boration choice.

Technical Reference(s): 1103-15B, Figure 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

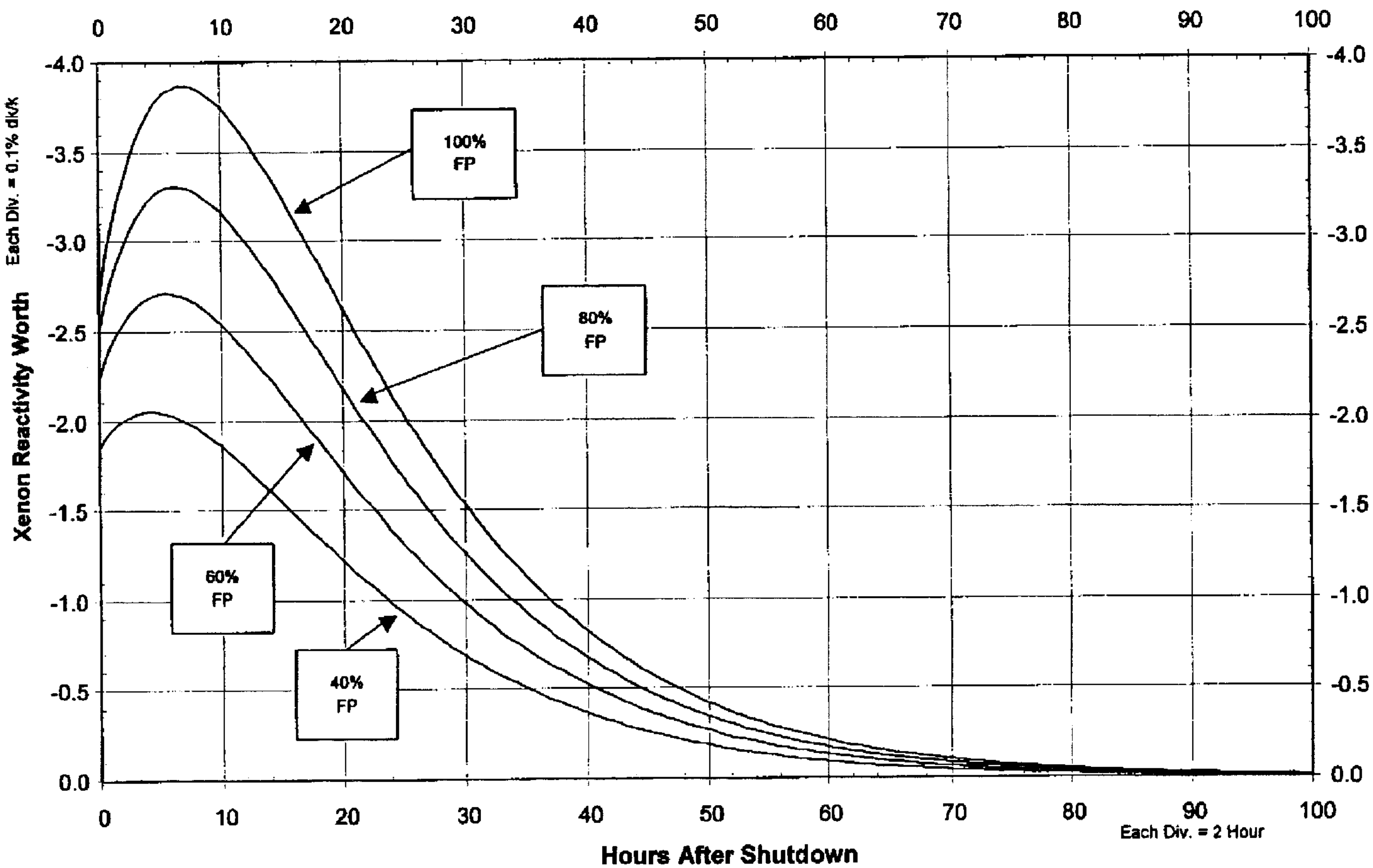
Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 b.1
55.43 ____________

Comments:

Figure 4
Cycle 16 Transient Xenon Reactivity Worth



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	K5.12 (011)	
	Importance Rating	2.7	3.3

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: Criteria and purpose of PZR level program

Proposed Question: Common 30

Which ONE of the choices correctly fills in the blanks of the following statement?

In accordance with OP-TM-220-000, "REACTOR COOLANT SYSTEM", with reactor power at 100% and MU-V-17 in HAND, Pressurizer Level must be maintained while operating _______ in order to prevent _______.

A. ≥ 200 inches; indication from going off-scale low following a reactor trip
B. ≥ 80 inches; heaters from de-energizing following a reactor trip
C. ≤ 385 inches; pressurizer safety valves from passing water if the turbine trips without a reactor trip
D. ≤ 260 inches; exceeding the Technical Specification pressurizer level limit following a turbine/reactor trip

Proposed Answer: A. ≥ 200 inches; indication from going off-scale low following a reactor trip

Explanation (Optional):

A. Correct – PZR level low limit at 100% Tave.
B. Plausible since that is a valid setpoint for heater cutoff but not the reason for minimum level during full power operation.
C. Plausible since it is the TS upper limit for criticality but the basis for that is to ensure a bubble in the PZR.
D. Plausible since it is the HI level alarm setpoint but level will respond in the direction opposite of the TS limit following a turbine/reactor trip.

Technical Reference(s): OP-TM-220-000, Reactor Coolant System, 2.2.9.2 (Page 9) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: _______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

2.2.7 Temperature differential between cold legs shall **not** exceed 5°F from 50 to 100 percent rated power with 4 RC pumps running. With less than 4 pumps running, the cold leg temperature differential limit is 10°F.

2.2.8 The fuel assemblies shall **not** be subjected to an external pressure above 2285 PSIG during normal steady state operation or above 2500 PSIG during transients.

2.2.9 Pressurizer level limits:

1. Absolute maximum pressurizer level at <u>any</u> time the reactor is critical is 385" (Technical Specification 3.1.3.4.1.)
2. To prevent pressurizer level from decreasing off scale following on a reactor trip, maintain pressurizer level (indicated) above the minimum level on Attachment 7.16 when the reactor is critical. When Tavg 579°F, then the minimum level is 200 inches.
3. The pressurizer must **not** be filled with water to solid water conditions (400 in.) at <u>any</u> time except as required for system hydrostatic tests.

2.2.10 Pressurizer maximum allowable heatup and cooldown rate is 100°F in <u>any</u> one hour. T.S. 3.1.2

2.2.11 To minimize chemistry differences between the RCS and Pressurizer, and to limit the temperature difference between the RCS and the spray & surge line piping, when the reactor coolant temperature is greater than 200°F, a minimum bypass spray flow of 0.75 gpm must be maintained.

3.0 REFERENCES

3.1 <u>220 System Procedures:</u>

3.1.1 OP-TM-220-201, IST of RC-V-1

3.1.2 OP-TM-220-202, IST of RC-V-2

3.1.3 OP-TM-220-203, IST PORV RC-V-2

3.1.4 OP-TM-220-204, IST of WDL-V-303 and WDL-V-304

3.1.5 OP-TM-220-205, IST of WDG-V-3 and WDG-V-4

3.1.6 OP-TM-220-206, Pressurizer Heaters Emergency Power Functional Test

3.1.7 OP-TM-220-241, IST of RC-V-1, RC-V-3 and RC-V-4

3.1.8 OP-TM-220-242, IST of RC-V-2 in Shutdown

3.1.9 OP-TM-220-243, IST of RC-V-40A/B, RC-V-41A/B, RC-V-28 and RC-V-44

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	K4.01 (017)	
Importance Rating	3.4	3.7

(K&A Statement) Knowledge of ITM system design feature(s) and/or interlock(s) which provide for the following: Input to subcooling monitors

Proposed Question: Common 31

Which one of the following identifies the reactor coolant temperature input to the Subcooling Margin Meter on PCL?

A. Narrow Range Thot.
B. Hottest In-Core Thermocouple.
C. Safety Grade Wide Range Thot.
D. Average of the 5 highest reading In-Core Thermocouples.

Proposed Answer: C. Safety Grade Wide Range Thot.

Explanation (Optional):

A. Plausible however it is the wide range instrument.
B. Plausible since it is likely to be the highest temperature.
C. Correct – temperature input per lesson plan.
D. Plausible since it is a monitored point.

Technical Reference(s): Lesson Plan TQ-TM-104-624-C001, Page 32 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: ____________

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.3
	55.43	

Comments:

2) Source Instruments are via MWT-1 and MWT-2 switches.

3) Selected indication on Control Board via MW 1, MW2 (Meters) (A5004)

4) The redundant process signals are monitored by SASS.

5) Both megawatt signals are recorded and the selected input is indicated.

5. Non-Nuclear Instruments NOT SASS Interfaced.

Other NNI that are of importance to monitoring plant parameters but are not SASS interfaced (i.e., do not provide controlled input for ICS), include:

Refer to OP-1105-6 and OP-1105-11 as applicable for additional data on these NNI.

E.g.,
o Monitored Parameter/Range
o Instrument #
o Location
o Uses/Function

a. <u>Safety Grade Wide Range RCS Pressure, 0 – 3000 psig: Rosemount Capacitance Type Detectors</u>

1) One detector per loop. PT 949, 963 (B and A loop, respectively)

2) Indication for "A" loop is on "A" RSP, indication for "B" loop is on the "thermo-hydraulic" section of panel PCL, and on the "B" RSP.

3) Both these PTs provide an input to its loop Saturation Margin Calculator and Meter.

4) RCS pressure input to DSS (Diverse Scram System)

5) PT-963 provides input to the PPC and is available on a loss of off-site power.

b. <u>Safety Grade Wide Range T_{hot}: 120°F - 920°F</u>

1) One detector per loop. TE 958, 960

2) Digital indication on PCL, analog readout on appropriate RSP.

3) Available on plant computer (A0838, A0843)

4) Provides input to Saturation Margin Calculator and Meter.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	K2.01 (041)	
	Importance Rating	2.8	2.9

(K&A Statement) Knowledge of bus power supplies to the following: ICS, normal and alternate power supply

Proposed Question: Common 32

Plant conditions:

- 100% power

Event:

- Loss of 1A ES MCC
- DC input breaker on 1E inverter trips.

Which ONE of the choices completes the following statement?

Power to ICS AUTO is from ________ and ICS HAND is _______.

A. TRA,
from TRB

B. TRA,
Lost

C. 1A DC via 1A inverter,
from TRB

D. 1A DC via 1A inverter,
Lost

Proposed Answer: C. 1A DC via 1A inverter,
from TRB

Explanation (Optional):

A. Plausible because TRA would be correct for 1A inverter failure, second part is correct.
B. Plausible because TRA would be correct for 1A inverter failure, second part plausible for not understanding static switch.
C. Correct because1A inverter swaps to DC, 1E inverter has lost both power supplies, static switch transfers to TRB.
D. Plausible because first part correct, second part plausible for lack of knowledge of static switch.

Technical Reference(s): DWG 206-051 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

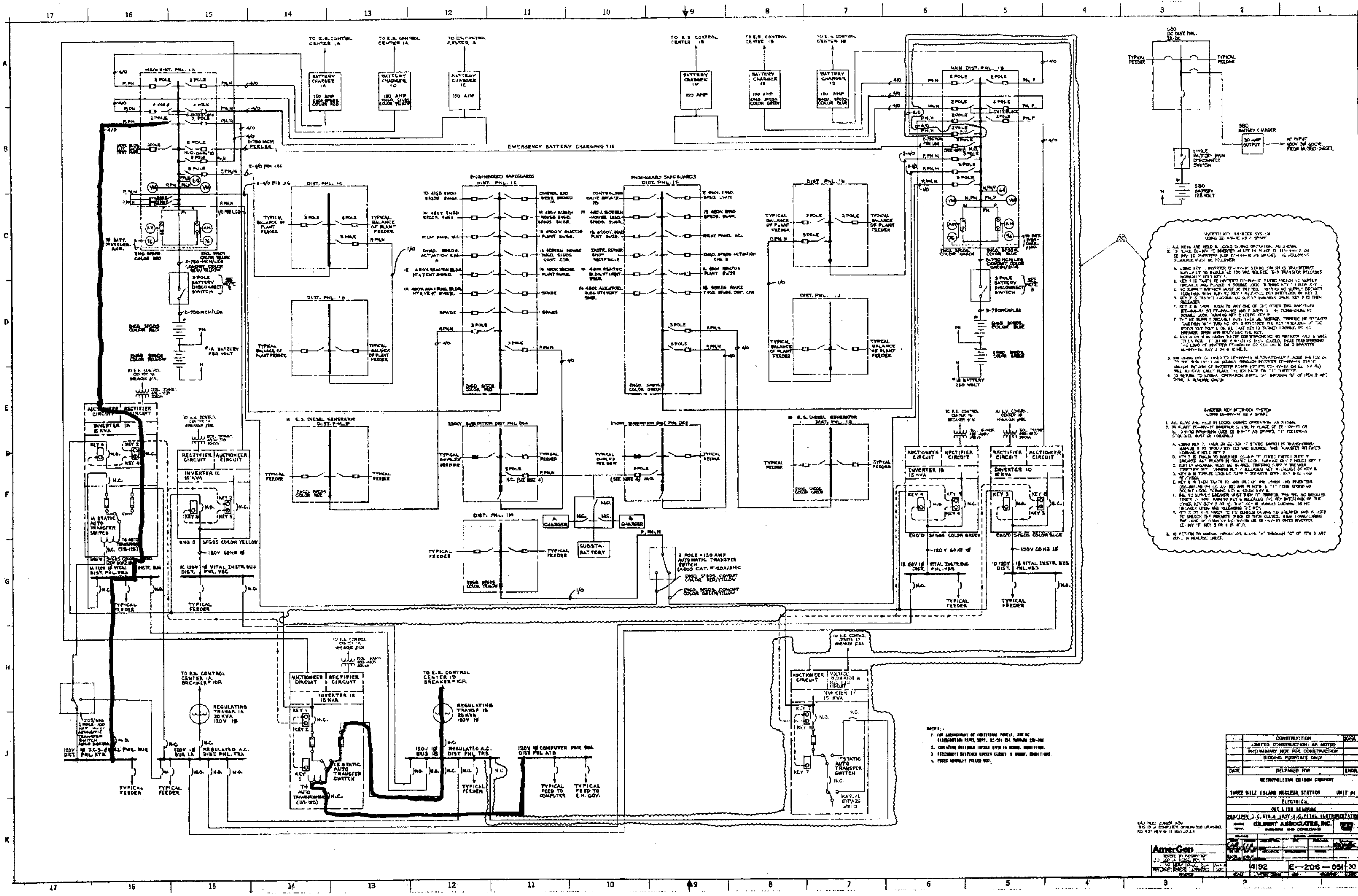
Question Source: Bank # ____________
Modified Bank # ____________ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 b.4
55.43 ____________

Comments:



MAIN DIST. PNL. 1A
MAIN DIST. PNL. 1B
EMERGENCY BATTERY CHARGING TIE
ENGINEERED SAFEGUARDS DIST. PNL. 1E
ENGINEERED SAFEGUARDS DIST. PNL. 1F
DIST. PNL. 1G
DIST. PNL. 1H
DIST. PNL. 1J
DIST. PNL. 1K
DIST. PNL. 1M
TO E.S. CONTROL CENTER 1A
TO E.S. CONTROL CENTER 1B
BATTERY CHARGER 1A
BATTERY CHARGER 1C
BATTERY CHARGER 1E
BATTERY CHARGER 1B
BATTERY CHARGER 1D
BATTERY CHARGER 1F
1A BATTERY 250 VOLT
1B BATTERY 250 VOLT
BATTERY DISCONNECT SWITCH
INVERTER 1A 15 KVA
INVERTER 1B 15 KVA
INVERTER 1C 15 KVA
INVERTER 1D 15 KVA
INVERTER 1E 15 KVA
RECTIFIER CIRCUIT
AUCTIONEER CIRCUIT
STATIC AUTO TRANSFER SWITCH
REGULATING TRANSF 1A 20 KVA 120V 1Φ
REGULATING TRANSF 1B 20 KVA 120V 1Φ
120V 1Φ COMPUTER PWR BUS DIST PNL ATB
TYPICAL FEED TO COMPUTER
TYPICAL FEED TO E.H. GOV.
TYPICAL FEEDER
TYPICAL BALANCE OF PLANT FEEDER
SUBSTA. BATTERY
A CHARGER
B CHARGER
3 POLE - 150 AMP AUTOMATIC TRANSFER SWITCH
MANUAL BYPASS
SBO BATTERY CHARGER
SBO BATTERY 125 VOLT
NOTES:-
CAD FILE: 206051 .K30
THIS IS A COMPUTER GENERATED DRAWING. DO NOT REVISE IT MANUALLY.
AmerGen
CONSTRUCTION
LIMITED CONSTRUCTION: AS NOTED
PRELIMINARY NOT FOR CONSTRUCTION
BIDDING PURPOSES ONLY
DATE
RELEASED FOR
ENGR.
METROPOLITAN EDISON COMPANY
THREE MILE ISLAND NUCLEAR STATION UNIT #1
ELECTRICAL
ONE LINE DIAGRAM
250/125V D.C. ETA. & 120V A.C. VITAL INSTRUMENTATION
GILBERT ASSOCIATES, INC.
ENGINEERS AND CONSULTANTS
4192
E—206—051
30

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	K4.02 (029)	
	Importance Rating	2.9	3.1

(K&A Statement) Knowledge of design feature(s) and/or interlock(s) which provide for the following: Negative pressure in containment

Proposed Question: Common 33

Plant conditions:

- Reactor is in cold shutdown on August 10, outside air temperature 85°F.
- Reactor Building personnel doors are open
- Reactor Building Purge is in progress in accordance with OP-TM-823-408, "RB Purge – RB Doors and/or Equipment Hatch Open"
- Both Purge Supply and both Purge Exhaust Fans are operating
- Air flow is continuously out of the RB doors

Which ONE of the following procedural actions must be taken?

A. Close AH-V-1C and AH-V-1D Purge Supply Valves.
B. Adjust RB Purge Manual Loader Purge Rate (AH-D-8B-EX1).
C. Secure one of the Purge Exhaust fans (AH-E-7A or AH-E-7B).
D. Throttle open Purge Exhaust Valve (AH-V-1B) to the 90° position.

Proposed Answer: C. ADJUST RB Purge Manual Loader Purge Rate (AH-D-8B-EX1).

Explanation (Optional):

A. Plausible because it changes flow rate but adjustments are done by repositioning dampers.
B. Correct – "if at any time" procedure step aligns with last bullet in stem.
C. Plausible because it changes flow rate but adjustments are done by repositioning dampers.
D. Plausible because it changes flow rate but the 30° travel limits must be removed to do this.

Technical Reference(s): OP-TM-823-408, RB Purge – RB Doors and/or Equipment Hatch Open (Page 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	824-GLO-10		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.9	
	55.43		

Comments:

NOTE: The objective is to obtain a minimal ΔP across RB personnel doors, and maintain flow into RB through any open hatch or personnel doors. If the flow into the RB is high, then RAISE purge supply flow (Open AH-D-8B). If the flow is continuously out of the RB then LOWER purge supply flow (Close AH-D-8B).

4.18 **IAAT** air flow at RB openings (door or hatch) is not flowing into RB with a minimal DP, **then ADJUST** RB Purge Manual Loader Purge Rate (AH-D-8B-EX1). ☐

4.19 **If** purge valve 30° travel limits have been removed, **then PERFORM** the following:

4.19.1 **HAND CRANK** AH-V-1C to full open 90° position. ____

4.19.2 **ENSURE CLOSE** AH-V-1B-BK, 1A ES VALVES MCC Unit 1B. ____

4.19.3 **CLOSE** AH-V-1C-BK, 1B ES VALVES MCC Unit 1B. ____

4.20 **MAKE** a page announcement that RB purge is in progress and normal access through RB openings is restored. ____

4.21 **MARK** FR-148 with start time, date, and release number. ____

4.22 **MONITOR** purge supply fan discharge temperature (AH-TI-6A and AH-TI-6B) at least once/shift. ____

4.23 **IAAT** one of the following conditions exists,

- Purge supply fan discharge temperature <u>**cannot**</u> be maintained ≥ 55°F,
- Average RB temperature below 320' el <u>**cannot**</u> be maintained ≥ 70°F,
- Purge flow is reduced to < 14,000 SCFM with both purge exhaust fans operating,

then GO TO Section 5.0 to stop RB purge. ☐

4.24 **INDICATE** purge rate obtained on Waste Gas Release Permit. ____

4.25 **If** restarting RB purge from a temporary shutdown, **then COMPLETE** release restart time information and reason for stop/restart on Waste Gas Release Permit. ____

4.26 **MAINTAIN** purge information as required on Waste Gas Release Permit. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	A2.03 (033)	
	Importance Rating	3.1	3.4

(K&A Statement) Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal Spent Fuel Pool water level or loss of water level.

Proposed Question: Common 34

Plant conditions:

- Reactor Coolant System is in Refueling Shutdown condition.
- The Fuel Transfer Canal has been filled.
- Fuel Transfer Canal isolation Valves FH-V-1A and FH-V-1B are Closed.

Event and subsequent conditions.

- Loss of Offsite Power (LOOP)
- Spent Fuel Pool Temperature is 180 °F, rising slowly.
- Spent Fuel Pool level is 342' and lowering.
- Alarm PLB-2-9, "Spent Fuel Pool A Level Lo", is illuminated.
- Alarm PLB-2-10, "Spent Fuel Pool B Level Lo", is illuminated.

Which ONE of the following actions is required in accordance with OP-TM-AOP-035, "Loss of Spent Fuel Cooling"?

A. Place both trains of Spent Fuel Pool Cooling in service to increase cooling.
B. Open FH-V-1A and FH-V-1B to equalize level and temperature with the refueling cavity.
C. Commence makeup to the Spent Fuel Pool from the BWST to restore boron concentration.
D. Initiate Attachment 1 for makeup to the Spent Fuel Pool from Fire Service Water to restore level.

Proposed Answer: D. Initiate Attachment 1 for makeup to the Spent Fuel Pool from Fire Service Water to restore level.

Explanation (Optional):

A. Plausible since both trains would be started if temperature is >190 °F pool level and pool level was not <343' 6".
B. Plausible since opening these valves would transfer some water to the Spent Fuel Pool; however the correct action is to close them.
C. Plausible since this is the procedure for transferring water to the Spent Fuel Pool; however it utilizes the RCBTs not the BWST transfer pumps not available in LOOP..
D. Correct – "if at any time level <343' 6 - - - "

Technical Reference(s): OP-TM-AOP-035, Loss of Spent Fuel Pool Cooling (Page 1 and 7) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

LOSS OF SPENT FUEL COOLING

1.0 ENTRY CONDITIONS

Either of the following conditions exist:

- Spent Fuel Pool temperature >160°F **and** rising.
- Spent Fuel Pool Level less than 343' 6" (PLB 2-9, PLB 2-10 alarm) **and** lowering.

2.0 IMMEDIATE ACTIONS – None

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Spent Fuel Pool temperatures are indicated on PPC points A0419 (A pool) and A0420 (B pool)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.1 **ANNOUNCE** the following: "Attention all personnel. A loss of Spent Fuel cooling has occurred. Non-essential personnel evacuate the Unit 1 Fuel Handling Building."	
☐ 3.2 **IAAT** spent fuel pool temperature >170°F, **or** Spent Fuel Pool level is < 343' 6" (low level alarm), **then INITIATE** Attachment 1, Spent Fuel Pool Makeup from Fire Service.	
☐ 3.3 **IAAT** spent fuel pool temperature >190°F, **then GO TO** Section 4.0.	
☐ 3.4 **IAAT** Spent Fuel Pool level is < 343' 6" (low level alarm) **then GO TO** Section 4.0.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
Unit Status Spent Fuel Pool temperature is >190°F or Spent Fuel Pool level can not be maintained.	
___ **4.0** **PLACE** SF-P-1A **and** SF-P-1B in Pull To Lock.	
___ 4.1 **If** spent fuel pool level is below the bottom of the Spent Fuel Pool level indicator (339' 11") and lowering **or** RM-G-9 radiation level is elevated and rising**, then** ____ **EVACUATE** fuel pool area. ____ **ACTIVATE** EPLAN. ____ **ENSURE** Fuel Handling Building Truck Bay Rollup Door is Closed. ____ **EXIT** this procedure.	
☐ 4.2 **WAAT** annunciator PLB-2-9, Spent Fuel Pool A Level Lo **or** PLB-2-10, Spent Fuel Pool B Level Lo alarms, **then RAISE** Spent Fuel pool level to the Hi alarm IAW 1104-29W, Transfers from the RCBTs to Spent Fuel Pools.	**PERFORM** Attachment 1, Spent Fuel Pool Makeup from Fire Service.
☐ 4.3 **IAAT** <u>all</u> of the following conditions exist: – Section 4.0 was entered due to loss of Spent Fuel Pool level – The leakage path has been isolated – Spent Fuel Pool level is > 343' 6" (low level alarm) **then GO TO** Section 5.0.	
___ 4.4 **VERIFY** the reactor is shutdown.	___ **If** spent fuel pool is boiling **or** pool level is not being recovered**, then INITIATE** plant shutdown to Hot Shutdown IAW 1102-4 and 1102-10.

ATTACHMENT 1
Spent Fuel Pool Makeup from Fire Service

Page 1 of 4

1.0 PURPOSE

This attachment provides instructions for addition of Fire Service water to the Spent Fuel Pools.

2.0 MATERIAL AND SPECIAL EQUIPMENT

2.1 Spent Fuel Pool fill assembly (pipe elbow, angle iron bracket & c-clamps to direct fire service into fuel pool) [FHB 305: outside SF Cooler Room AOP Box #2]

2.2 2 spanner wrenches (for removing hose sections)

3.0 PRECAUTIONS, LIMITATIONS, AND PREREQUISITES

3.1 Precautions

3.1.1 Personnel should be aware that the spent fuel pool area will be a hot and humid environment.

3.2 Limitations – None

3.3 Prerequisites

3.3.1 Spent Fuel Pool level is above the bottom of the Spent Fuel Pool level indicator (339' 11").

4.0 MAIN BODY

4.1 Perform the following to align the Fire System to supply water to the Spent Fuel Pools: ____

4.1.1 **CLOSE** FS-V-60 (AB 281' 2' East of elevator 7' above floor). ____

4.1.2 **VERIFY** FS-V-143 is Closed (FHB 348' in Spent Fuel Pool area). ____

4.1.3 **REMOVE** nozzle **and** any sections of collapsible hose if present, from fire hose on hose reel at FS-V-143. ____

4.1.4 **ATTACH** temporary fuel pool fill assembly to fire hose. ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	A4.01 (045)	
Importance Rating	3.1	2.9

(K&A Statement) Ability to manually operate and/or monitor in the control room: Turbine valve indicators (throttle, governor, control, stop, intercept), alarms, and annunciators.

Proposed Question: Common 35

Plant conditions:

- The reactor is critical.
- Operators are starting up the main turbine generator in accordance with OP-TM-301-102, "MAIN TURBINE GENERATOR STANDBY MODE TO OPERATING MODE".
-

Which ONE of the choices fills in the blank in the following statement?

In preparation for High Pressure Turbine Shell warming, the operator performs the step "SELECT ON and Execute Command for Shell Warming", _______ will OPEN.

A. all Control Valves
B. all Stop Valves
C. all Intercept Control Valves
D. all Intermediate Stop Valves

Proposed Answer: A. all Control Valves

Explanation (Optional):

A. Correct – procedure step 4.9.6 verifies control valves open.
B. Plausible since these valves open later in this procedure when turbine roll is selected.
C. Plausible since these valve do control steam to the low pressure hoods; however the LP hoods are not heated during this evolution.
D. Plausible since these valves do open when the turbine is reset; however they close when shell warming is selected.

Technical Reference(s): OP-TM-301-102, Main Turbine Generator Standby Mode to Operating Mode (Page 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

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

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ________

Comments:

NOTE: Speed error >4 rpm caused by turbine rolling during shell warming will cause the CVs to close partially to maintain zero speed.

4.10.4 **SELECT** #2 Screen, Chest/Shell Warming screen. ____

4.10.5 **SELECT** ON **and** Execute Command for Shell Warming. ____

4.10.6 **VERIFY** the following occurs:

- Intercept control valves remain Closed. ____
- Intermediate stop valves go Closed. ____
- Control valves Open fully. ____

NOTE: The SV-2 stem will move approximately 0.05% for each Open/Close button operation. Total available opening stroke for warming is 10.3%. Shell pre-warming plot (main menu) may be used to monitor warming rates and progress.

NOTE: Turbine shell pressures may be monitored using:

- Comp Pt. A0037, Main Turb 1st STE Press A
- Comp Pt. A0038, Main Turb 1st STE Press B
- Gauge panel on 355' elevation TB
- #2 Chest & Shell Warming screen

NOTE: Alarms K-1-6 on DTCS Process Alarm #552 may be caused by shell warming. No action is required.

CAUTION
First stage shell inner surface metal temperature warming rate limit is 150°F/hr (Comp. Pt. C4131) Turb HP Shell Inner Heatup Rate.

CAUTION
Monitor OSTG Pressure and TBV demand to ensure steam supply is sufficient, **not** to reduce RCS temperature < 525°F.

4.10.7 **THROTTLE** SV-2 to raise HP shell pressure to between 60 **and** 100 PSIG using either of the following methods: ____

- Set Position feature.
- Using the Open/Close pushbuttons.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	A3.02 (071)	
	Importance Rating	2.8	2.8

(K&A Statement) Ability to monitor automatic operation of the Waste Gas Disposal System including: Pressure-regulating system for the waste gas vent header.

Proposed Question: Common 36

Plant conditions:

- Makeup Tank venting is required to adjust gas concentration.

Which ONE of the following is the required Waste Gas Compressor control alignment for the Makeup Tank venting evolution?

A. One Waste Gas Compressor must be in HAND.
B. Both Waste Gas Compressors must be in HAND.
C. One Waste Gas Compressor must be in AUTOMATIC.
D. Both Waste Gas Compressors must be in AUTOMATIC.

Proposed Answer: A. One Waste Gas Compressor must be in HAND.

Explanation (Optional):

A. Correct. One WDG-P is selected to HAND prior to venting.
B. Plausible since one is placed in HAND and the system load increases.
C. Plausible since this is a normal alignment.
D. Plausible since this implies availability to handle the system load increase.

Technical Reference(s): OP-TM-211-483, Venting MU-T-1 (Page 2) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

4.0 MAIN BODY

4.1 **If** MU-T-1 O_2 concentration > 2 percent, **then VERIFY** vent header H_2 concentration is < 4% ____

4.2 **If** MU-T-1 H_2 concentration > 4 percent, **then VERIFY** vent header O_2 concentration is < 2% ____

4.3 **If** venting flow rate adjustments are expected, **then POSITION** operator in communication with control room at MU-V-65. ____

4.4 **VERIFY CLOSED** MU-V-28 (CC). ____

4.5 **CLOSE** WDG-V-65. ____

4.6 **START** WDG-P-1A **or** 1B in HAND (RW Panel). ____

4.7 **PERFORM** the following while Venting is in progress:

4.7.1 **MONITOR** MU17PI (CC). ____

4.7.2 **If** venting flow rate is excessive (greater than 2 psig / minute) **then**,

1. **THROTTLE** MU-V-65 to obtain 2 psig/minute. ____
2. **ENSURE** tamper seal is replaced on MU-V-65. ____

4.8 **VENT** MU-T-1 to desired pressure using MU-V-13 (CC). (Ref Attachment 7.3, OP-TM-211-000) ____

DATE	PRINTED NAME	SIGNATURE	(INITIAL ONE)	
			INITIALS	IV/CV INITIALS

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	K3.02 (072)	
	Importance Rating	3.1	3.5

(K&A Statement) Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Fuel handling operations

Proposed Question: Common 37

Plant conditions:

- Refueling is in progress.

Event:

- RM-G-9, Fuel Handling Building (FHB), fails full scale.
- Technicians have confirmed that it is an electronic component failure.
- Repairs will take two hours.
- Fuel movement was halted when the channel failed.

Which ONE of the following describes the status of fuel handling operations with RM-G-9 failed?

A. Fuel handling activities can resume as long as RM-A-4, FHB Exhaust, remains operable.
B. Fuel handling activities can resume after RM-A-14, FHB ESF Ventilation, is verified in service.
C. Spent Fuel Bridge fuel handling and refueling operations are halted until portable instrumentation is substituted for RM-G-9.
D. Irradiated fuel movement into the FHB must remain suspended until RM-G-9 is restored to operability.

Proposed Answer: C. Spent Fuel Bridge fuel handling and refueling operations are halted until portable instrumentation is substituted for RM-G-9.

Explanation (Optional):

A. Plausible because RM-A-4 and RM-G-9 perform the same interlock function.
B. Plausible because it is normally aligned to the FHB during refueling.
C. Correct - TS 3.8.1 allows substitution of portable instruments.
D. Plausible because RM-G-9 or a substitute is required.

Technical Reference(s): TS 3.8.1, Page 3-44 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.11
55.43 ______

Comments:

3.8 FUEL LOADING AND REFUELING

Applicability: Applies to fuel loading and refueling operations.

Objective: To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the Reactor Building refueling area shall be monitored by RM-G6 and RM-G7. Radiation levels in the spent fuel storage area shall be monitored by RM-G9. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one decay heat removal pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the Reactor Building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the Reactor Building at least one door in each of the personnel and emergency air locks shall be capable of being closed.* The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

------------------------------ NOTE ------------------------------

The equipment hatch may be open if all of the following conditions are met:

1) The Reactor Building Equipment Hatch Missile Shield Barrier is capable of being closed within 45 minutes,
2) A designated crew is available to close the Reactor Building Equipment Hatch Missile Shield Barrier, and
3) Reactor Building Purge Exhaust System is in service.

3.8.7 During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or capable of being closed,* or

2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

* Administrative controls shall ensure that the Reactor Building Purge Exhaust System is in service, appropriate personnel are aware that air lock doors and/or other penetrations are open, a specific individual(s) is designated and available to close the air lock doors and other penetrations as part of a required evacuation of containment. Any obstruction(s) (e.g., cable and hoses) that could prevent closure of an air lock door or other penetration will be capable of being quickly removed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	A4.06 (086)	
	Importance Rating	3.2	3.2

(K&A Statement) Ability to manually operate and/or monitor in the control room: Halon system

Proposed Question: Common 38

Plant conditions:

- 100% power.

Event:

- The Air Intake Tunnel North Tunnel (AIT) Halon unit has actuated.

Which ONE of the following describes plant response?

A. All three deluge systems actuate and ONLY Fire Pumps FS-P-2 and FS-P-3 start on interlock with the actuation.
B. All three deluge systems actuate and ONLY Fire Pump FS-P-1 starts based on header pressure lowering to the setpoint.
C. The associated deluge system actuates and ONLY Fire Pump FS-P-1starts on interlock with the actuation.
D. The associated deluge system actuates and ONLY Fire Pumps FS-P-2 and FS-P-3 start based on header pressure lowering to the setpoint.

Proposed Answer: A. All three deluge systems actuate and ONLY Fire Pumps FS-P-2 and FS-P-3 start on interlock with the actuation.

Explanation (Optional):

A. Correct – per note on Page 6 of the reference.
B. Plausible because the first part is correct.
C. Plausible because first part would match zone and pump start on interlock is correct.
D. Plausible because the second part is correct.

Technical Reference(s): OP 1104-45I, Air Intake Tunnel Halon System (Page 6) (Attach if not previously provided)

1104-45D (Page 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

Automatic Fire Pump Start

The actuation of the Deluge System listed below closes contacts in the Motor Driven Fire Pump (FS-P-2) and the River Water Diesel Fire Pump (FS-P-3) starting circuits causing the pumps to start.

1. Auxiliary Transformer 1A
2. Auxiliary Transformer 1B
3. Main Transformer 1A and 1B
4. Turbine Building East Wall
5. Hydrogen Seal Oil Unit
6. Turbine Oil Reservoir and Oil Conditioner
7. Feedwater Pumps Turbine Oil Reservoir and Conditioner
8. Diesel Generator Cooling Air Intakes
9. Diesel Generator A Combustion Air Intake
10. Diesel Generator B Combustion Air Intake
11. Air Intake Tunnel Vertical Inlet Section
12. Air Intake Tunnel Inside Curtain Wall
13. Air Intake Tunnel Upstream
14. Control Building Supply Air Filter AH-F-3A/3B
15. Reactor Building Purge Exhaust Air Filter AH-F-1
16. Aux. Building Exhaust Air Filter AH-F-2A/2B/2C/2D
17. Control Building Area Filter AH-F-10/11

4.0 EQUIPMENT OPERATION

4.1 Normal Operation - **LEVEL 3**

> **NOTE**
>
> The AIT Halon and Deluge system normal operation is an armed state, awaiting an actuation signal. Entry into the AIT requires defeat of system per Section 4.2.

4.1.1 Prerequisites:

A. FS in service per 1104-45B

B. Halon in service per 1004-45I

4.1.2 Procedure

A. **MONITOR** PL-A, PL-B, H&V A/B for alarms (see Table 45I-2).

B. **REFER** to appropriate alarm responses and evaluate system operability.

> **NOTE**
>
> Actuation of any AIT Protective device will isolate AIT and stop affected fans to contain the hazardous condition. Refer to alarm responses, 1104-19 and 1104-15A as needed.

> **NOTE**
>
> Actuation of any of the Halon systems will cause an actuation of all three deluge systems and auto start fire pumps. Refer to alarm responses, 1104-45J, 1104-45B, and 1104-45D as appropriate.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EK2.02 (007)	
	Importance Rating	2.6	2.8

(K&A Statement) Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects

Proposed Question: Common 39

The Diverse Scram System (DSS) provides backup protection for an ATWS (Abnormal Transient Without Scram) by _____________ the shunt trip coils on the _____________ 480V breakers.

A. energizing; CRD #10 and #11
B. de-energizing; CRD #10 and #11
C. de-energizing; 1L-2A and 1G-2A (CRD System Feeders)
D. energizing; 1L-2A and 1G-2A (CRD System Feeders)

Proposed Answer: D. energizing; 1L-2A and 1G-2A (CRD System Feeders)

Explanation (Optional):

A. Plausible because the first part is correct.
B. Plausible because these breakers do have shunt trips but neither part of the answer is correct.
C. Plausible because second part is correct
D. Correct answer. Shunt trip energizes to actuate and the correct breakers are identified.

Technical Reference(s): Lesson Plan TQ-TM-104-641-C001, Reactor Protection System, Page 9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: 641-GLO-2 (As available)

Question Source: Bank # IR-641-GLO-2-Q01

Modified Bank # _____ (Note changes or attach parent)

New _____

Question History: Last NRC Exam 5/2003

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

10 CFR Part 55 Content: 55.41 b.6
55.43

Comments:

Content/Skills	Activities/Notes
3. Components external to the RPS cabinets include:	
a. Manual Trip pushbutton.(Console Center)	
b. CRD UV coil contacts.(at CRD breakers 338' CB)	
4. The DSS components include the following:	
a. DSS cabinets (338' CB Patio)	
1) Inside: Trip Testing/Bypass Switches	
2) Outside: Status lamps	
b. DSS Manual trip button (Console Center)	
c. DSS shunt trip coils (322' CB Patio at 1G-2A and 1L-2A breakers)	
d. RCS pressure input instrumentation.	
1) RC-PT-949	
2) RC-PT-963	
. Explanation The Reactor Trip Module is the heart of the RPS channel. The other modules will be covered in later sections of this lesson plan.	**This section is highly detailed and not required for ILT/NLO operators. It is more appropriate for requal students who are familiar with the RPS system.**
1. Reactor Trip Module The reactor trip modules are located in the NI/RPS cabinets. Each module controls a separate circuit breaker and/or contactor in the Control Rod Drive Control System. Each reactor trip module receives four trip signals; one from its associated RPS channel and one from each of the other three RPS channels. 120 VAC power is provided to the CRDs through this module, which forms a 2-out-of-4 trip logic.	**ILT students should understand that removing a module from a trip string can initiate a trip signal and that removing a reactor trip module, even in manual bypass, will cause a trip signal.**
Separate lights are provided on the face plate of the reactor trip module to indicate whether a trip action is due to a real condition or on line testing. Whenever a trip action does occur, the main relay must be manually reset, using a toggle switch mounted on the module face plate. The main sections that make up the reactor trip module are: • Manual Bypass,	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AA2.17 (008)	
	Importance Rating	2.5	2.7

(K&A Statement) Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Steam dump valve controller (position)

Proposed Question: Common 40

Plant conditions:

- The reactor tripped due to a leak through Pressurizer Code Safety Valve RC-RV-1A.
- The CRS initiated EOP-001, "Reactor Trip", then EOP-002, "Loss of Subcooling Margin (SCM)", then EOP-006, "LOCA Cooldown".
- Incore Thermocouple Temperature is 522 °F.
- RCS pressure is 975 psig.
- Core cooldown rate is 42 °F/hr.
- EOP-010, Rule 1, "Loss of Subcooling Margin (SCM)", has been completed.
- EOP-010, Guide 6, "OTSG Pressure Control", has been ordered initiated by the CRS.

Which ONE of the following is the next action to be taken?

A. Reduce OTSG pressure to obtain a 100°F/hr cooldown rate.
B. Maintain OTSG pressure more than 100 psig below RCS Pressure.
C. Reduce OTSG pressure such that secondary Tsat is 40-60 °F lower than Incore Thermocouple Temperature.
D. Maintain OTSG pressure such that secondary Tsat is 90-100 °F lower than Incore Thermocouple Temperature.

Proposed Answer: B. Maintain OTSG pressure more than 100 psig below RCS Pressure.

Explanation (Optional):

A. Plausible since this is the maximum normal cooldown rate desired. However, Guide 6 requires lowering OTSG Pressure more than 100 psig below RCS Pressure when SCM is <25 °F.
B. Correct answer. Step 1 of Guide 6 RNO column.
C. Plausible since this is a step in EOP-006 if cooldown rate is not adequate and primary to secondary heat transfer does not exist with RCS temperature above 300 °F.
D. Plausible since this is a step in EOP-006 if cooldown rate is not adequate and primary to secondary heat transfer does not exist with RCS temperature above 300 °F, and, lowering Tsat to 40-60 °F lower than Incore Thermocouple temperature did not work.

Technical Reference(s): OP-TM-EOP-006, LOCA Cooldown (Page 3) (Attach if not previously provided)

OP-TM-EOP-010, Guide 6, OTSG Pressure Control

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.10

55.43 ______

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 3.9 **VERIFY** G1-02 **and** G11-02 are CLOSED.	____ **If** EG-Y-1B is operating **or** in ES standby, **then INITIATE** OP-TM-861-901 to place EG-Y-1A in ES standby ____ **If** EG-Y-1A is operating **or** in ES standby, **then INITIATE** OP-TM-861-902 to place EG-Y-1B in ES standby
____ 3.10 **INITIATE** Rule 5, "Emergency Boration."	
☐ 3.11 **IAAT** BWST level < 15 ft, **or** RB flood level > 54", **then BRIEF and INITIATE** Guide 21, "Transfer to RB Sump Re-circulation."	
☐ 3.12 **IAAT** LPI flow > 1250 GPM in each line, **then GO TO** Step 3.26.	
☐ 3.13 **IAAT** <u>all</u> of the following exist ___ Primary-to-secondary heat transfer does **not** exist ___ RCS > 300F ___ Core cooldown rate < 40°F/hr, **then GO TO** Section 4.	
____ 3.14 **ENSURE** OTSG pressure is being controlled IAW Guide 6.	

Guide 6
OTSG Pressure Control

IAAT the reactor is shutdown, **then**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** SCM > 25°F.	1. **MAINTAIN** OTSG pressure more than 100 psig below RCS pressure. 2. **END**
2. **VERIFY** RCS cooldown is **not** required.	1. **ADJUST** OTSG pressure to control desired cooldown rate within limits per Guide 11. 2. **END**
3. **VERIFY** RCS leakage is **not** causing a RCS cooldown.	1. **ADJUST** OTSG pressure to control desired cooldown rate within limits per Guide 11. 2. **END**
4. **ADJUST** OTSG pressure to stabilize T_{COLD} **and** maintain OTSG pressure less than 1020 psig.	
5. **VERIFY** all MSSVs are closed.	**LOWER** OTSG pressure to attempt to reseat the safety valve. **ENSURE** pressure reduction does not cause an excessive cooldown rate or Tcold < 525°F

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	G2.1.2 (009)	
	Importance Rating	3.0	4.0

(K&A Statement) Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: Common 41

Plant conditions:

- A small break LOCA resulted in a reactor trip and 1600 psig ESAS actuation.
- The operating crew is performing EOP-006, "LOCA COOLDOWN".
- Tave is 450 ˚F.

Which ONE of the following would require immediate operator action?

A. RCS cooldown rate is 40 ˚F/hour with all Reactor Coolant Pumps stopped.
B. Pressurizer level has exceeded 80 inches with HPI in operation.
C. BWST level is 20 feet and lowering slowly.
D. RB pressure has exceeded 4 psig.

Proposed Answer: D. RB pressure has exceeded 4 psig.

Explanation (Optional):

A. Plausible since this exceeds value for < 255 ˚F.
B. Plausible since this would permit implementation of plant cooldown procedure if HPI were stopped.
C. Plausible since this is an IAAT step but the value is 15 feet.
D. Correct answer (IAAT Statement/Carryover Step).

Technical Reference(s): EOP-006, Step 3.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

LOCA COOLDOWN

1.0 ENTRY CONDITIONS

When directed by EOP.

2.0 IMMEDIATE ACTIONS - None

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ TIME	
____ 3.1 **ENSURE** HPI and LPI are operated IAW Rule 2.	
☐ 3.2 **IAAT** an ES Actuation setpoint is reached, **then ENSURE** all ESAS components have actuated.	____ **INITIATE** contingency actions IAW Section 4.2 of the applicable procedure(s). - OP-TM-211-901 "Emergency Injection" - OP-TM-214-901 "RB Spray Operation" - OP-TM-244-901 "Containment Isolation" - OP-TM-534-901 "RB Emergency Cooling"
____ 3.3 **ENSURE** CF-V-1A **and** CF-V-1B are OPEN. (PCR)	
☐ 3.4 **IAAT** RM-A-1 high alarm **or** 4 psig ESAS have actuated, **then INITIATE** OP-TM-826-901, "Control Building Ventilation System Radiological Event Operations."	
☐ 3.5 **IAAT** RCS > 25°F superheat, **then GO TO** EOP-008.	
____ 3.6 **INITIATE** Guide 20, "PRIOR to Transfer to RB Sump."	
____ 3.7 **ENSURE** performance of an alarm review.	
____ 3.8 **REQUEST** SM evaluate Emergency Action Levels (EALs).	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AK2.07 (015)	
	Importance Rating	2.9	2.9

(K&A Statement) Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP seals

Proposed Question: Common 42

Plant conditions:

- 35% reactor power and rising after a brief maintenance outage.
- Reactor Coolant Pump RC-P-1A #1 Seal leakoff has been steadily lowering for the past hour and is now .7 gpm.
- MAP F-1-3, "RCP SEAL #1 LEAKOFF HI/LO", is actuated.
- The Reactor Operator reports that RC-P-1A bearing water temperature has started to rise.

Which one of the following is the required action?

A. Immediately trip the reactor; trip RC-P-1A; close MU-V-33A when RC-P-1A stops rotating.
B. Immediately trip RC-P-1A; close MU-V-33A when RC-P-1A stops rotating.
C. Within 5 minutes: Trip the reactor; trip RC-P-1A; place RC-P-1A in the standby mode.
D. Within 5 minutes: Trip RC-P-1A; place RC-P-1A in the standby mode.

Proposed Answer: D. Within 5 minutes: Trip RC-P-1A; place RC-P-1A in the standby mode.

Explanation (Optional):

A. Plausible because it is the correct action for excessive RC-P seal leakoff with power > limit for < 4 pumps.
B. Plausible because it is the correct action for excessive RC-P seal leakoff with power < limit for < 4 pumps.
C. Plausible because it is the correct action for inadequate RC-P seal leakoff with power > limit for < 4 pumps.
D. Correct – "if at any time" action (low flow-rising temperature) in alarm response procedure.

Technical Reference(s): OP-TM-MAP-F0103, Page 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	531-GLO-5	(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 b.10	
	55.43	

Comments:

4.0 (Continued)

4.2 **If** Seal Number 1 Leak-Off Flow (SLO) is < 0.8 gpm,
then PERFORM the following:

NOTE: Excessive Number 2 seal leakage may be indicated by Lo # 1 SLO, High RCDT Inleakage, possible RB Sump accumulation, and possible elevated vibrations due to vibration sensor on RCP Seal package.

- **IAAT** Seal Number 1 Leak-Off Flow (SLO) is < 0.8 gpm,
  and RCP has been operating > 24 hours,
  then PERFORM OP-TM-226-150 series procedure to place affected RCP in the Standby mode.
- **IAAT** Seal Number 1 Leak-Off Flow (SLO) is < 0.8 gpm,
  and RCP Seal or Bearing Water Temperatures are rising (Ref: OP-TM-PPC-A0521),
  then within **5** minutes **SECURE** affected RCP as follows:
  - **ENSURE** reactor power is < RPS Limit for final RCP Combination,
    or TRIP the Reactor IAW OP-TM-EOP-001.
  - **TRIP** affected RCP (CC).
  - **PERFORM** OP-TM-226-150 series procedure to place affected RCP in standby mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AA2.02 (022)	
	Importance Rating	3.2	3.7

(K&A Statement) Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Charging pump problems

Proposed Question: Common 43

Plant conditions:

- Reactor power 100%.
- MU-P-1B is OOS for maintenance.

Event:

- Reactor trip due to Loss of Offsite Power (LOOP).
- EG-Y-1A and EG-Y-1B energize their 1D and 1E 4160 Volt ES busses respectively.
- 1P 480 Volt ES Bus de-energizes due to an electrical fault.
- Makeup Tank level is in the restricted operating region of the curve.

Which one of the following Makeup Pump – BWST suction valve combinations can be aligned to provide makeup flow?

A. Start MU-P-1A and open MU-V-14A.
B. Start MU-P-1A and open MU-V-14B.
C. Start MU-P-1C and open MU-V-14A.
D. Start MU-P-1C and open MU-V-14B.

Proposed Answer: D. Start MU-P-1C and open MU-V-14B.

Explanation (Optional):

A. Plausible because 4160 V power is available but power is not available to the oil pumps.
B. Plausible because the second part is correct.
C. Plausible because MU-P-1C is available but no power available to MU-V-14A.
D. Correct – power is available to MU-P-1C, supporting equipment and MU-V-14B.

Technical Reference(s): OP 1107-4 Electrical Distribution Panel Listing (Pages 46, 47, 51) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
- 55.41 b.7
- 55.43 ______

Comments:

480V CONTROL CENTER

1A ENGINEERED SAFEGUARDS

(B-201-043)

UNIT	EXACT WORD DESIGNATOR	SYMBOL	ELEM. PRINT	CONT. FUSE
6B	Main Turbine Oil Lift Pump LO-P-7C/D	LO-P-7C/D	208-545	ABC 1
6C	Main Turbine Oil Lift Pump LO-P-7E/F	LO-P-7E/F	208-545	ABC 1
6D	Main Turbine Turning Gear GN-Y-1 and Piggy Back GN-Y-1/PBM	GN-Y-1	208-537	ABC 3
7A	Control Building Return Fan A AH-E-19A	AH-E-19A	208-614	ABC 1
7B	Control Building Chilled Waterpump A AH-P-3A	AH-P-3A	208-642	ABC 2
7C	Cont Bldg Emer Vent Supply Fan AH-E-18A	AH-E-18A	208-613	ABC 2
7D	Borated Wtr Stor Tank Outlet Val. MU-V-14A	MU-V-14A	208-440	ABC 1
7EL	Emer Diesel Gen 1A Aux Panel (480 Vac)	EE-PNL-EGAUX-1A		
7ER	Feeder Xfmr Unit 8B	CT-5		
8A	Dist Panel CT-5 120-208V	CT-5 Panel		
8B	Transformer 480/120-208V			
9A	Cond Line Isolation Valve CO-V-14A	CO-V-14A	208-475	ABC 1
9B	Pressurizer Spray Valve RC-V-1	RC-V-1	208-488	ABC 1
9C	Spare			ABC 1
9D	Space Not Available			
10A	Spare			
10B	Spare			
10C	Normal Vent Supply Fan AH-E-17A	AH-E-17A	208-612	ABC 3
10D	Spare			
11A	Intermediate Clg Closed Loop Pump A IC-P-1A	1C-P-1A	208-527	ABC 3, FNQ-3
11B	Penetration Cooling Fan AH-E-9A	AH-E-9A	208-617	ABC 3
12AL	Rad Monitoring RMA4 Pump	RMA4 Pump		
12AR	Rad Monitoring RMA9 Pump	RMA9 Pump		
12BL	Rad Monitoring RMA6 Pump	RMA6 Pump		
12BR	Lighting Panel CT-1 (Normal)	CT-1		
12C	Air Cooling Fan A Emer FW Pump AH-E-24A	AH-E-24A	208-670	ABC 2
12D	Condensate Storage Tank Tie CO-V-111A	CO-V-111A	208-505	OTM 1
12E	Main Turb Turn Gear Oil Pump LO-P-5	LO-P-5	208-542	ABC 2
13A	ES Load Shedding Relays R1A & R2A "ind. light"	(Location Only - No Breaker in this Cubicle)		
13B	RC-P-1A High Press Oil Lift Pump RC-P-2A-1	RC-P-2A-1	208-530	ABC-1

LOCATION Control Tower Elev 322 East Room

FED FROM 1P 480V Engineered Safeguards SWGR

480V CONTROL CENTER

1A ENGINEERED SAFEGUARDS VALVES

(B-201-052)

UNIT	EXACT WORD DESIGNATOR	SYMBOL	ELEM. PRINT	CONT. FUSE
1A	NaOH Tank Outlet Valve BS-V-2A	BS-V-2A	208-502	ABC 1
1B	RB Purge Exhaust Isol Valve AH-V-1B	AH-V-1B	208-494	ABC 1
1C	DH Pump Discharge Valve DH-V-4A	DH-V-4A	208-433	ABC 2
1DL	ESF Unit 1A Heater AH-C-57A	AH-C-57A	209-976	
1DR	Heat Trace for BS-T-2A	HT-BS-T-2A		
2A	BS Pump Suction BS-V-3A	BS-V-3A	208-504	ABC 1
2B	Main Feedwater Block Valve FW-V-5A	FW-V-5A	208-425	ABC 2
2C	FW-V-92A (FW-V-16A Upstream Isolation Valve)	FW-V-92A	208-524	ABC 1
2D	Makeup Pump Recirc Isolation Valve MU-V-36	MU-V-36	208-690	ABC 1
3A	Borated Wtr Stor to DH Pump A Valve DH-V-5A	DH-V-5A	208-432	ABC 1
3B	Reactor Bldg Sump to DH Pump A Valve DH-V-6A	DH-V-6A	208-434	ABC 1
3C	DH Pump Discharge to Makeup Pump DH-V-7A	DH-V-7A	208-431	ABC 1
3D	RC-P-1A ICCW Isolation Valve IC-V-79A Brkr. open at power for Appendix R concerns	IC-V-79A	208-512	ABC 1
4A	Makeup Pump B Aux. Oil Pump MU-P-2B	MU-P-2B	208-523	ABC 1
4B	Emergency Makeup Valve MU-V-16A	MU-V-16A	208-442	ABC 1
4C	Emergency Makeup Valve MU-V-16B	MU-V-16B	208-442	ABC 1
4D	Rx Coolant Pump Seal Return Isol Valve MU-V-25	MU-V-25	208-441	ABC 1
5A	Core Flood Tank A Sample Valve CF-V-2A	CF-V-2A	208-445	ABC 1
5B	Core Flood Tank B Sample Valve CF-V-2B	CF-V-2B	208-445	ABC 1
5C	BS Pump Discharge Valve BS-V-1A	BS-V-1A	208-501	ABC 1
5D	RCP Stand Pipe Fill Valve MU-V-39	MU-V-39	208-691	ABC 1
6AL	Lighting Panel AB-1 (Emergency)	AB-1 (Emer)		
6AR	480V Receptacles A1, A7	A1, A7		
6BL	Feeder Heat Tracing Panel 3A-1	HT-PNL-3A1		
6BR	Feeder Heat Tracing Panel 3A-2	HT-PNL-3A2		
6C	MU PP A Aux Oil PP MU-P-2A	MU-P-2A	208-523	ABC 1
6D	Reactor Bldg Vent Header Isol Valve WDG-V-3	WDG-V-3	209-316	ABC 1
6E	Steam Generator A Sample Isol Valve CA-V-4A	CA-V-4A	209-356	ABC 1

LOCATION Aux Bldg North End Near Radwaste Panel

FED FROM 1P 480V Engineered Safeguards SWGR

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AA1.01 (025)	
	Importance Rating	3.6	3.7

(K&A Statement) Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS/RHRS cooldown rate

Proposed Question: Common 44

Plant conditions:

- Cooldown in progress using the "A" Decay Heat Removal (DH) Train.
- Decay Heat Return Temperature 185 °F.
- Cooldown rate 12 °F/hr.
- All Reactor Coolant Pumps are secured.

Event:

- DH Pump 1A trips.
- The "B" DH Removal train has been started in accordance with OP-TM-212-211, "Shifting DHR Train B From DHR Standby To DHR Operating Mode".
- Decay Heat Return Temperature rose to 189 °F during the transfer.
- Decay Heat Return Temperature lowered from 189 °F to 174 °F when the "B" train was started.

Which ONE of the following is the allowable action?

A. Hold RCS temperature constant for a minimum of 30 minutes using In-Core Thermocouple temperature indication.
B. Hold RCS temperature constant for a minimum of 30 minutes using Decay Heat Return Temperature Indicator DH-2-TI1.
C. The cooldown can continue to ≥ 159 °F within the next 30 minutes using In-Core Thermocouple temperature indication.
D. The cooldown can continue to ≥ 159 °F within the next 30 minutes using Decay Heat Return Temperature Indicator DH-2-TI1.

Proposed Answer: B. Hold RCS temperature constant for a minimum of 30 minutes using Decay Heat Return Temperature Indicator DH-2-TI1.

Explanation (Optional):

A. Plausible since the first part is correct.
B. Correct answer. The 15 °F step change is the limit for the 30 minutes time period.
C. Plausible since this is the limit if the 15 °F step change is ignored.
D. Plausible since this is the limit if the 15 °F step change is ignored. The correct indicator is used.

Technical Reference(s): TS Figure 3.1-1, Reactor Coolant System Heat-up/Cooldown Limitations (Attach if not previously provided)

OP-1102-11, Plant Cooldown (Page 13, 36)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

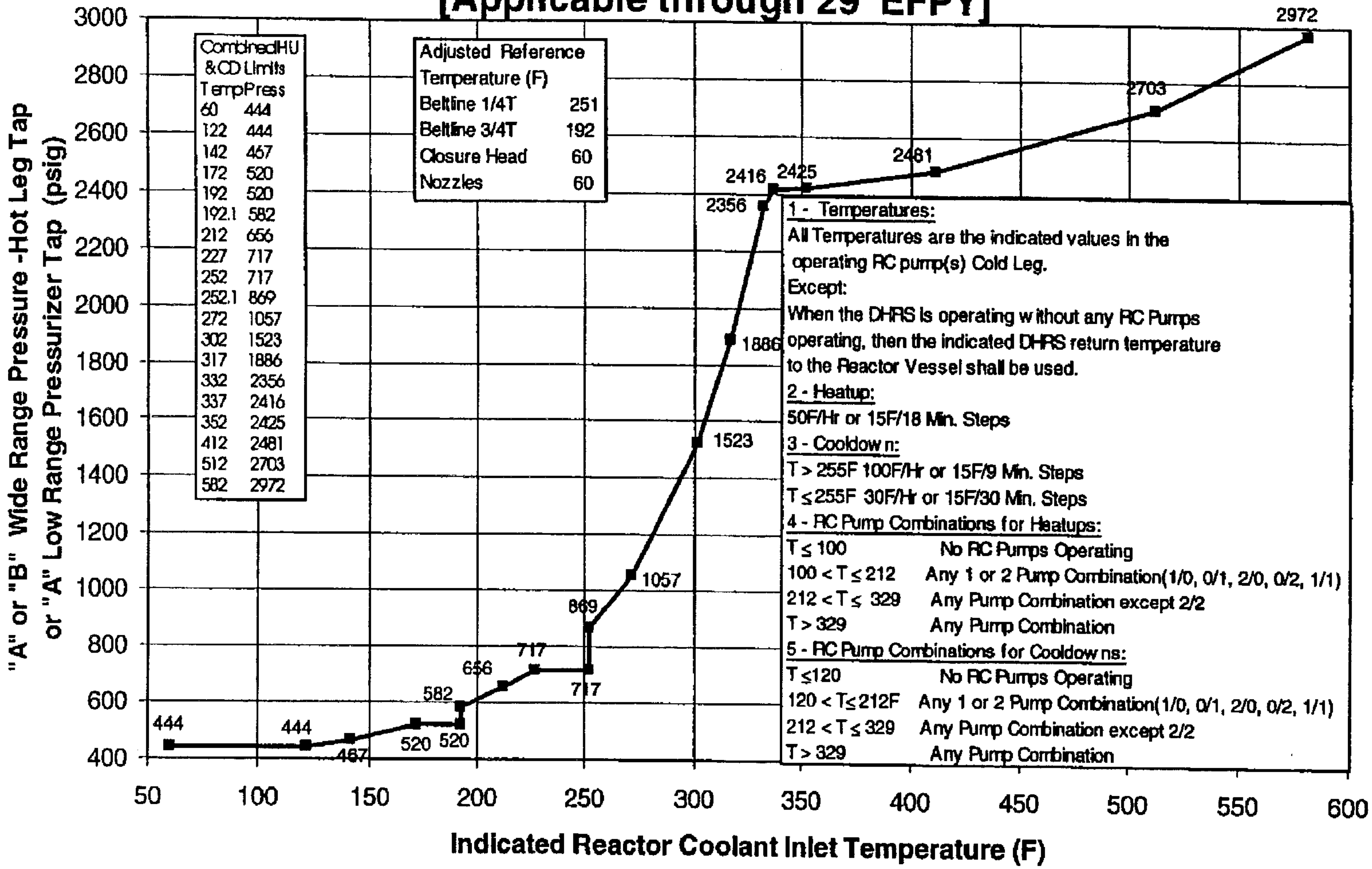
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.10

55.43 ______

Comments:

Figure 3.1-1
Reactor Coolant System Heatup/Cooldown Limitations
[Applicable through 29 EFPY]



RCS Press/Temp and Cooldown Monitoring Requirements

NOTE
These requirements are for compliance with Tech. Spec. 3.1.2.

1. Every 30 minutes, **PLOT** a point **and RECORD** the time on the P/T curves (Enclosure 4 Figure 1 or 1A). **If** the plant computer calculated cooldown rates are unavailable **then COMPLETE** Enclosure 4 Data Sheet every 5 minutes to verify RCS, and PZR cooldown/heatup rates are within limits.

2. **USE** the following preferred instruments **and RECORD** which instrument is used on top of the data sheet (a comparable instrument may be used if these instruments are unavailable).

   Time: use the clock on PLF

RCS Pressure:	at < 450 psig use RC3A-PT5/PI2 (PLF) 450 - 1700 psig use "WR A Pressure" on RC3-PR (CC) or PI-949A (PCL) At > 1700 psig use "NR A Pressure" on RC3-PR (CC) or PI-949A (PCL)
RCS Temperature:	With RCPs operating use RC-TI-959A or RC-TI-961A(PCL When RCPs are shutdown, while on DHR use DH2-TI1 or 2
Pressurizer Temp:	Determine the saturation temperature for A Hot Leg WR RC Pressure using steam tables.
RCS Cooldown Rate:	With RCPs operating, use C4000 (5 min. avg.) & C4002 (1 min. avg.) and record C4039 (30 min. avg.) When RCPs are shutdown, while on DHR use C4041 (5 min. avg.)
PZR Cooldown Rate:	Use point C4158 (1 min. avg.) and C4159 (5 min avg.) and record C4040 (30 min. avg.)

3. Cooldown Rate Limits

   When RCS temperature is > 255°F: **MAINTAIN** cooldown rate less than 100°F / Hr. (or 15°F / 9 minutes).

   When RCS temperature is $\leq$ 255°F: **MAINTAIN** cooldown rate less than 30°F / Hr. (or 15°F / 30 minutes).

   Pressurizer cooldown rate limit is 100°F in any one hour.

4. Every 30 minutes , **RECORD** the values for RCS pressure, temperature and pressurizer temperature on the data sheets.

5. Every 30 minutes, **CALCULATE** cooldown rates from data recorded on Data Sheet as follows:

$$\frac{\text{Temp (2)} - \text{Temp (1) °F}}{\text{Time(2)} - \text{Time (1) min}} \times 60 = \_\_\_\_\_\_\_\_\_\_ \text{ °F/HR}$$

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	AK3.03 (026)	
Importance Rating	4.0	4.2

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW/nuclear service water

Proposed Question: Common 45

Plant conditions:

- The unit was at 100% power when a steam line break caused a reactor trip and ESAS actuation.

Which ONE of the following is the reason for the procedural action to verify the trip and lock out the non-ES selected Nuclear Services River Water Pump?

A. to limit load on the 1R and 1T 480 Volt ES buses due to Decay Heat River Water Pump power requirements.
B. to limit load on the A and B Emergency Diesel Generators during a loss of offsite power (LOOP) with an ESAS signal present.
C. to prevent excessive cooling of the Nuclear Services Closed Cooling Water System during and ESAS condition since heat load is less than during normal operation.
D. to prevent pump damage if only two Nuclear Services Closed Coolers are in service when the ESAS signal is received.

Proposed Answer: A. to limit load on the 1R and 1T 480 Volt ES buses due to Decay Heat River Water Pump power requirements.

Explanation (Optional):

A. Correct answer. Changes to the DR-P-1A/B motors required limiting the number of NR pumps running on the 1R and 1T 480 Volt ES buses during and ESAS condition.
B. Plausible since this would limit the load on the Emergency Diesels; however it is not a concern for diesel loading.
C. Plausible since the heat load is less during and ESAS condition where the RCPs are secured; however the third pump running would not have a significant affect on Nuclear Services Closed Cooling water temperature and it could be started by defeating the ESAS signal and resetting the 86 lockouts if a running pump were to trip.
D. Plausible since it is possible to have the coolers throttled during normal operation; however the third pump running would not have a significant affect on pump or cooler operation.

Technical Reference(s): TQ-TM-104-531, Primary Cooling Systems (Page 37, GLO-5)

OP-TM-861-902 4.3.5 sub step 1

Proposed references to be provided to applicants during examination: ______

Learning Objective: 541-GLO-5 (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

d) Remote readout / alarm in control room (PRF) and PPC (computer).

4. Interlocks

a. NSRW Interlocks

1) Kinney Automatic Strainer

If in auto, it is interlocked to start when Nuclear River Pump is started, and to stop when Nuclear River pump is stopped.

Also, strainer backwash valve (if in auto) will open on pump start signal.

2) Standby pump auto-start

If a running pump trips, standby (non-running) pump auto-starts.

If preferred STBY pump does not start in 0.5 seconds, other STBY Pump will start, if not running. Sequence is A-B-C-A.

Auto start blocked if an ES condition is present in order to reduce load on 1R & 1T 480v Busses due to power requirements for DR-P-1A/1B

3) ESAS actuation

a) ES selected NS pumps start on Block 3.

(1) 43 SS - Determines which pump is selected for ES

(2) A or B Pump selected on 1 R BUS

(3) B or C Pump selected on 1T BUS

b) Non-ES Selected NR pump will trip on an ES actuation.

c) NR-V-4A/B Close on ES signal

4) Kirk Key Interlock for NR-P-1B BKRS on 1R and 1T buses

a) Prevents having two power supplies for one component energized simultaneously (prevents paralleling 1R and 1T buses).

4.3. <u>WHILE EG-Y-1B is loaded (UNIT ops) on 1E 4160V bus</u>

4.3.1. **SHUTDOWN** non essential ES bus loads to maintain EG-Y-1B load < 3.0 MWe. ____

4.3.2. **IAAT** additional loads will be added to B train ES bus**, then** ☐

- **USE** Attachment 1 **and VERIFY** the steady state EDG load will remain < 3.0 Mwe.
- Limit 1S 480V bus current to less than 185 amps.

4.3.3. **ADJUST** governor to maintain frequency between 59 and 61 Hz. ____

4.3.4. **IAAT** 1E 4160V bus steady state voltage is NOT between 4100 and 4300 Volts, **then**:

1. **ADJUST** Unit Voltage Rheostat (DG B: Inside local alarm panel) (Key #11)

CAUTION
There is no design feature for bumpless transfers of Voltage control between AUTO and MANUAL. Do not return voltage control to AUTO.

2. **IF** AUTO voltage control is not functional, **then PLACE** Exciter control in MANUAL **and ADJUST** manual voltage controller to maintain 4160V bus voltage between 4100 and 4300 volts.

4.3.5. **IAAT ESAS actuates, then** ☐

1. **VERIFY** the proper response of all ESAS actuated components. ____
2. **MATCH** flags and amber disagreements for affected ES components. ____
3. **OBTAIN** US concurrence **and BYPASS** the ESAS signal. ____
4. **RESET** BUS 1S **and** BUS 1T Lockout relays. ____

4.3.6. **IAAT** jacket coolant temperature (A0375) > 195°F **or** lube oil temperature (A0373) > 223°F**, then REDUCE** EDG load to maintain temperature within these limits. ☐

4.3.7. **If** AH-E-29B is **<u>not</u>** available, **then INITIATE** OP-TM-861-911,"Emergency Ventilation of EG-Y-1B Room". ____

4.3.8. **ENSURE** AH-E-29B is operating. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EK1.01 (029)	
	Importance Rating	2.8	3.1

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior

Proposed Question: Common 46

Plant conditions:

- Reactor power 100%.

Event:

- Both Main Feedwater Pumps trip.
- The automatic reactor trip fails.
- The automatic turbine trip fails.

Which ONE of the choices completes the following statement describing the response of Tcold and nuclear instrumentation if the reactor continues untripped?

Tcold will ______ and Power Range Nuclear Instrumentation will indicate ________ than actual Core Thermal Power.

A. Lower,
   Higher
B. Lower,
   Lower
C. Rise,
   Higher
D. Rise,
   Lower

Proposed Answer: C. Rise,
Higher

Explanation (Optional):

A. Plausible as OTSGs dry out pressure drops, in a wet generator Tcold follows Tsat, 2nd part plausible for misconception of downcomer temperature affects on NI's.
B. Plausible as OTSGs dry out pressure drops, in a wet generator Tcold follows Tsat, 2nd part is correct.
C. Correct answer. as OTSGs dry out heat transfer stops, 2nd part rising Tcold allows more leakage, NIs indicate higher.
D. Plausible 1st part correct, 2nd part plausible for misconception of downcomer affect on NIs..

Technical Reference(s): TQ-TM-104-EOP001, Pg. 7 (Attach if not previously provided)
OS-24, 3.16 (Pg. 6)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.5
55.43 ______

Comments:

PURPOSE:

Verify reactor shutdown has occurred or is in progress.
The response not obtained guidance provides action for a failure of the reactor to trip (i.e. ATWS or Anticipated Transient Without Scram).

BASIS:

Verification of reactor shutdown (i.e. reactor power < 7% on the power range Nuclear Instruments) ensures emergency feedwater and atmospheric dump valves are capable of providing adequate heat removal. The value of 7% is indicative that the prompt drop in neutron flux has occurred. It also corresponds to the reactor protection system (RPS) anticipatory trip on loss of feedwater.

Nuclear instrumentation can indicate erroneous levels if a LOCA occurs due to voiding in the vessel and increased neutron leakage observed by the excore detection system. For a normal reactor trip with groups 1-7 fully inserted, the excore detection system will immediately indicate a prompt drop in flux level, followed by a decay eventually equal to a –80 second period.

Breakers 1L-02 and 1G-02 deenergize the power supplies for the rod control system and are included as preferred action for the unlikely failure of both the reactor trip and DSS pushbuttons. If additional breaker failures are encountered, OS-24 provides in "Contingency Actions" the authority to deenergize necessary power supplies to ensure control rod insertion.

The Anticipated Transient Without Scram (ATWS) action serves to balance heat production with heat removal. Maintaining a balanced primary to secondary heat transfer rate prevents an excessive RCS temperature and pressure transient while the reactor is critical. If main feedwater is not available, then this balance cannot be maintained, the main turbine is tripped and EFW is initiated to provide a source of OTSG heat transfer to mitigate the heat transfer imbalance. For the (ATWS) with a loss of

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	EK3.03 (038)	
Importance Rating	3.6	4.0

(K&A Statement) Knowledge of the reasons for the following responses as the apply to the SGTR: Automatic actions associated with high radioactivity in S/G sample lines

Proposed Question: Common 47

Plant conditions:

- 100% power.
- Overhead alarm MAP C-1-1, "RADIATION LEVEL HI", just actuated.
- The source of the alarm is RM-A-5 CONDENSER VACUUM PUMP EXHAUST.

Which ONE of the choices completes the following statement?

The MAP-5 Sampler starts to sample the....

A. steam lines for iodine when RM-A-5 reaches the ALERT setpoint.
B. steam lines for iodine and tritium when RM-A-5 reaches the ALERT setpoint.
C. condenser offgas for iodine when RM-A-5 reaches the ALARM setpoint.
D. condenser offgas for iodine and tritium when RM-A-5 reaches the ALARM setpoint.

Proposed Answer: C. condenser offgas for iodine when RM-A-5 reaches the ALARM setpoint.

Explanation (Optional):

A. Plausible since the sampler will start on RM-A-5 ALARM and samples for iodine but not that point.
B. Plausible since the sampler does sample for Condenser Offgas iodine; the tritium sample is not associated with RM-A-5 setpoints but is sampled on Offgas.
C. Correct – per associated lesson plan.
D. Plausible since the sampler starts on RM-A-5 alarm and samples for iodine. The tritium sample is collected from the offgas line but is not associated with the MAP-5 Sampler.

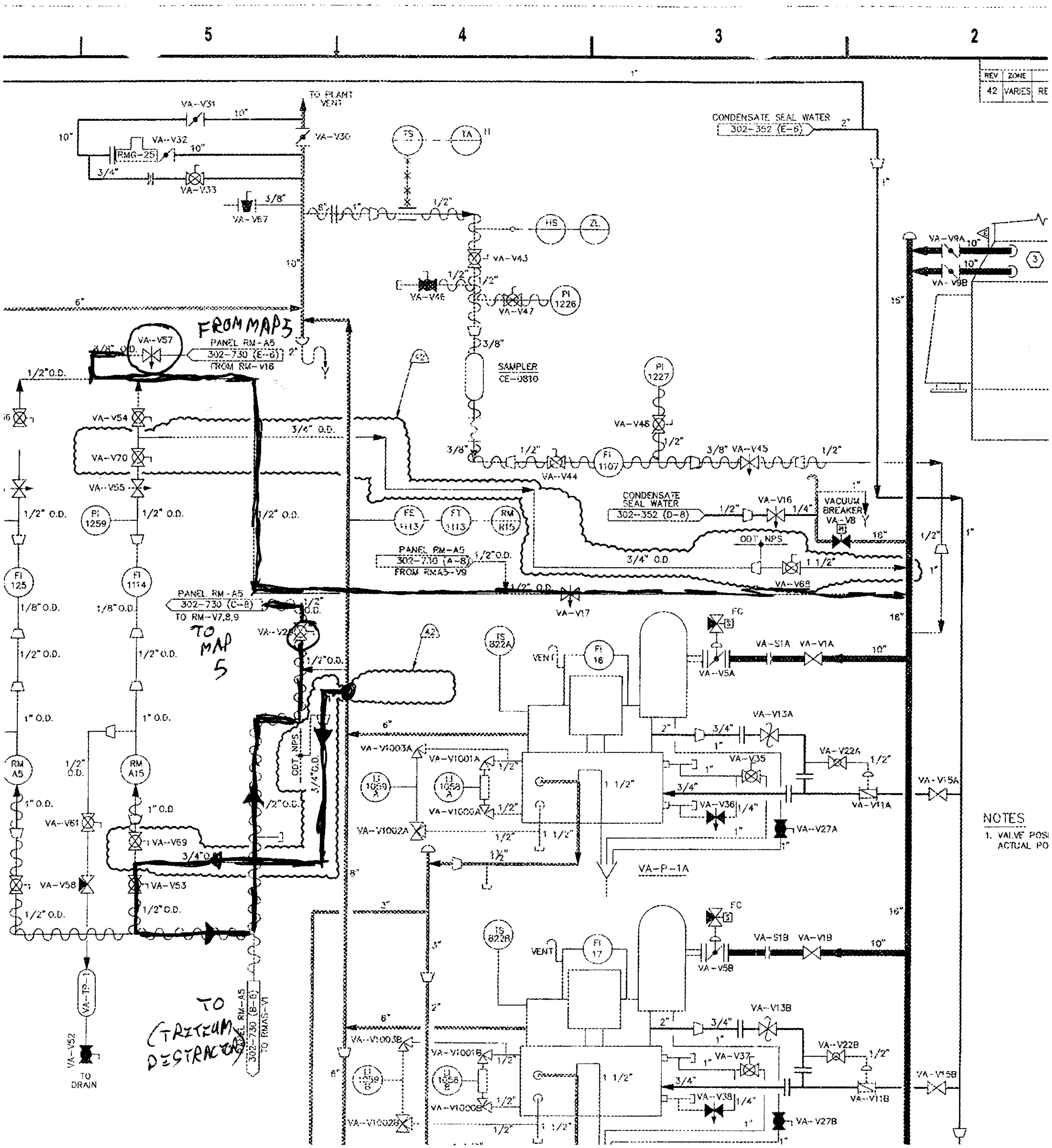
Technical Reference(s): 1105-8 Table 3.7 step 3.7.4 (Attach if not previously provided)
DWGs 302-131 & 302-730

Proposed references to be provided to applicants during examination: None

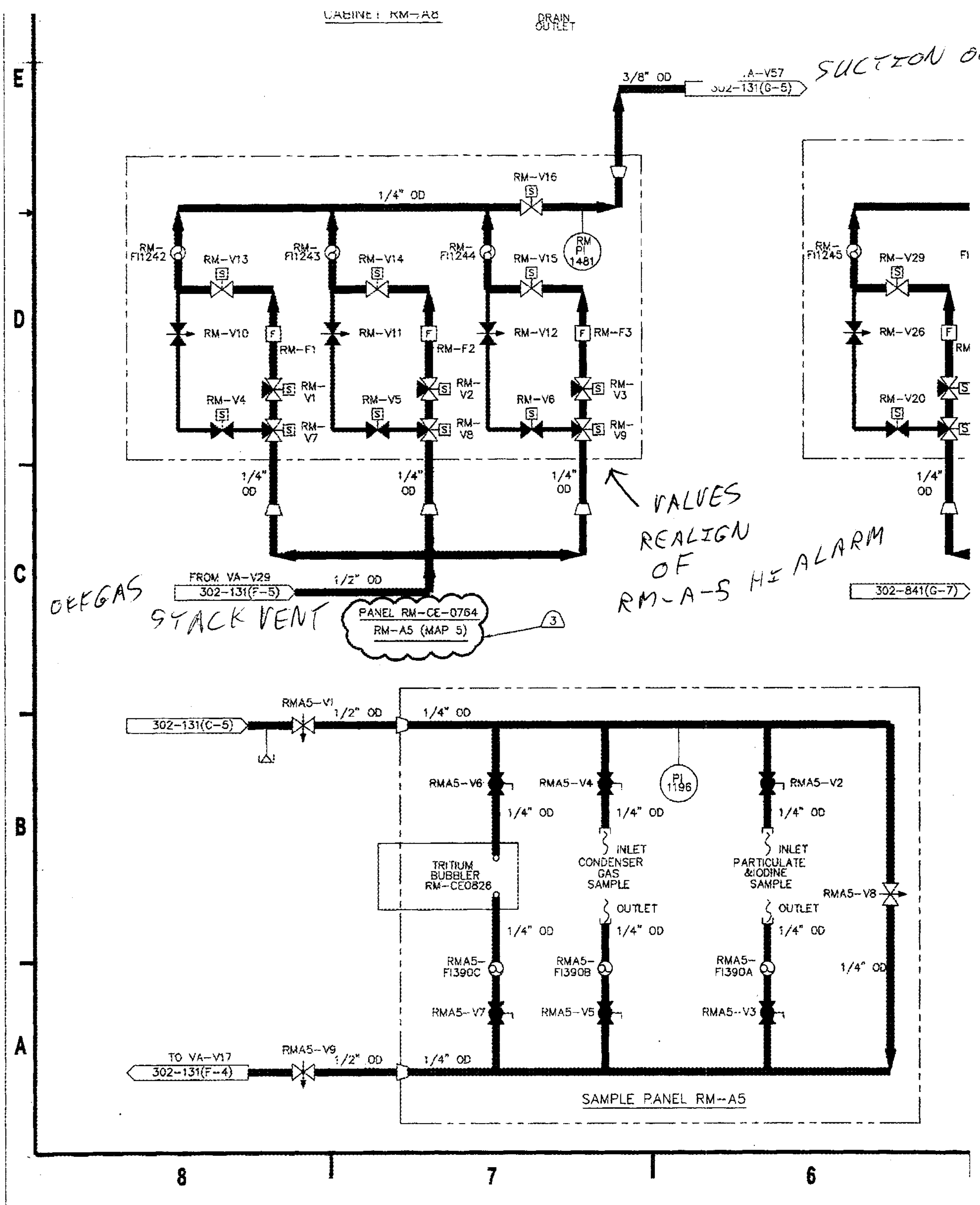
Learning Objective: 661-GLO-2 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.11	
	55.43		

Comments:


5
4
3
2
REV ZONE
42 VARIES
TO PLANT VENT
VA-V31
VA-V32
RMG-25
VA-V33
VA-V30
VA-V67
CONDENSATE SEAL WATER
302-352 (E-6)
TS
TA
HS
ZL
VA-V43
VA-V46
PI 1226
VA-V47
FROM MAP 5
VA-V57
PANEL RM-A5
302-730 (E-6)
FROM RM-V16
SAMPLER
CE-0810
PI 1227
VA-V48
VA-V45
VA-V44
FI 1107
VA-V9A
VA-V9B
VA-V54
VA-V70
VA-V55
PI 1259
FI 125
FI 1114
FE 1113
FT 1113
RM R15
CONDENSATE SEAL WATER
302-352 (D-8)
VA-V16
VACUUM BREAKER VA-V8
ODT NPS
PANEL RM-A5
302-730 (A-8)
FROM RMA5-V9
VA-V68
VA-V17
PANEL RM-A5
302-730 (C-8)
TO RM-V7,8,9
TO MAP 5
VA-V28
TS 822A
FI 16
VA-S1A
VA-V1A
VA-V5A
VA-V13A
VA-V22A
VA-V35
VA-V36
VA-V11A
VA-V27A
VA-V15A
VA-V1003A
VA-V1001A
VA-V1000A
VA-V1002A
VA-P-1A
RM A5
RM A15
VA-V61
VA-V69
VA-V58
VA-V53
VA-TP-1
VA-V52
TO DRAIN
TO (TRITIUM DESTRACTOR)
PANEL RM-A5
302-730 (B-6)
TO RMA5-V1
TS 822B
FI 17
VA-S1B
VA-V1B
VA-V5B
VA-V13B
VA-V22B
VA-V37
VA-V38
VA-V11B
VA-V27B
VA-V15B
VA-V1003B
VA-V1001B
VA-V1000B
VA-V1002B
NOTES
1. VALVE POSI
ACTUAL PO

FROM
302-131


CABINET RM-A8
DRAIN OUTLET
SUCTION OF VACUUM PUMPS
3/8" OD
VA-V57
302-131(G-5)
1/4" OD
RM-V16
RM PI 1481
RM-FI1242
RM-FI1243
RM-FI1244
RM-FI1245
RM-V13
RM-V14
RM-V15
RM-V29
RM-V10
RM-V11
RM-V12
RM-V26
RM-F1
RM-F2
RM-F3
RM-V1
RM-V2
RM-V3
RM-V4
RM-V5
RM-V6
RM-V20
RM-V7
RM-V8
RM-V9
VALVES
REALIGN
OF
RM-A-5 HI ALARM
FROM VA-V29
302-131(F-5)
1/2" OD
OFFGAS
STACK VENT
PANEL RM-CE-0764
RM-A5 (MAP 5)
3
302-841(G-7)
302-131(C-5)
RMA5-V1
RMA5-V6
RMA5-V4
PI 1196
RMA5-V2
TRITIUM BUBBLER RM-CE0826
INLET
CONDENSER GAS SAMPLE
OUTLET
PARTICULATE &IODINE SAMPLE
RMA5-V8
RMA5-FI390C
RMA5-FI390B
RMA5-FI390A
RMA5-V7
RMA5-V5
RMA5-V3
TO VA-V17
302-131(F-4)
RMA5-V9
SAMPLE PANEL RM-A5
E
D
C
B
A
8
7
6

FROM 302-730

TABLE 3.7
Atmospheric Radiation Monitors

Step No.	Aux & FHB Vent Radiation Monitor (RM-A-8 and Gas Hi Channels)	Aux. Bldg. Vent Radiation Monitor (RM-A-6)	FHB Vent Radiation Monitor (RM-A-4)	RB Purge Radiation Monitor (RM-A-9 and Gas Hi and RM-G-24)	Condenser Off Gas Radiation Monitor (RM-A-5, RM-A-15 and RM-G-25)	Waste Gas Release Radiation Monitor (RM-A-7)
3.7.1 TS /ODCM	TS table 3.5-3 ODCM Part 1 Table 2.1-2 item 5	ODCM Part 1 Table 2.1-2 item 5	ODCM Part 1 Table 2.1-2 item 5	TS Table 3-5.1 item C.3.f TS Table 3.5-3 items 1d and 1e TS 3.8.9 ODCM Part 1 Table 2.1-2 item 3	TS table 3.5-3 ODCM Part 1 Table 2.1-2 item 4	ODCM Part 1 Table 2.1-2 item 1
3.7.2 Redundant Instruments & Compensatory Actions	RM-A-4 <u>AND</u> RM-A-6	RM-A-8	RM-A-8	No redundant monitor.	RM-A-5 or 15 (2 channels required)	No redundant monitor.
3.7.4 Interlock	If RM-A-8G Hi alarm is actuated, then interlock TRIPS AH-E-10 & 11 CLOSES WDG-V-47 STARTS RM-A-8 MAP 5.	If RM-A-6G Hi alarm is actuated, then interlock TRIPS AH-E-11.	If RM-A-4G Hi alarm is actuated, then interlock TRIPS AH-E-10. CLOSES AH-D-120, 121 & 122.	If RM-A-9G Hi Alarm is actuated, then interlock CLOSES AH-V-1A,B,C,D CLOSES WDL-V-534, 535. STARTS RM-A-9 MAP 5.	No DEFEAT req'd If RM-A-5 Hi Alarm is actuated, then interlock STARTS RM-A-5 MAP 5.	If RM-A-7 Hi alarm is actuated, then interlock CLOSES WDG-V-47.
3.7.5 Shutdown	**PLACE** each of the following switches in OFF: _____ RM-A-8P _____ RM-A-8I _____ RM-A-8G _____ RM-A-8G Hi	**PLACE** each of the following switches in OFF: _____ RM-A-6P _____ RM-A-6I _____ RM-A-6G	**PLACE** each of the following switches in OFF: _____ RM-A-4P _____ RM-A-4I _____ RM-A-4G	**PLACE** each of the following switches in OFF: _____ RM-A-9P _____ RM-A-9I _____ RM-A-9G _____ RM-A-9G Hi _____ RM-G-24	**PLACE** each of the following switches in OFF: _____ RM-A-5 _____ RM-A-15 _____ RM-G-25	**PLACE** each of the following switches in OFF: _____ RM-A-7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AK3.03 (054)	
	Importance Rating	3.8	4.1

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Manual control of AFW flow control valves

Proposed Question: Common 48

Plant conditions:

- Plant has tripped due to a Loss of Offsite Power (LOOP).
- No equipment was out of service prior to the trip.
- 1D & 1E ES 4160V buses are powered by their respective Emergency Diesel Generator.
- SCM is 47 °F, decreasing slowly
- OTSGs are at 30% on operating range and lowering.
- OTSG A/B pressure is 1010 psig and stable.
- Incore Temperatures are rising.

Which ONE of the following is the reason for placing the Emergency Feedwater Control Valves (EF-V-30's) in MANUAL?

A. To feed the OTSGs at the maximum available rate because incore temperatures are rising.
B. To allow feed rate to be adjusted to maintain OTSG levels at ≥ 25" in the STARTUP Range.
C. To lower feed rate to ≥ 215 gpm per OTSG since subcooling requirements are met.
D. To raise OTSG level to 75-85% in the OPERATING Range because no RCP is running.

Proposed Answer: A. To feed the OTSGs at the maximum available rate because incore temperatures are rising.

Explanation (Optional):

A. Correct – Rule 4, B.2 RNO step.
B. Plausible since this is the correct value for an RCP running.
C. Plausible since it is an RNO for subcooling > 25 °F.
D. Plausible since this is the rate is correct but for subcooling < 25 °F.

Technical Reference(s): OP-TM-EOP-010, Rule 4, Feedwater Control (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	424-GLO-10		(As available)
Question Source:	Bank #		
	Modified Bank #	QR-424-GLO-10-Q04	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

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. **VERIFY** SCM > 25°F.	**If** both OTSGs are available **and** OTSG tube leak < 1 gpm, **then FEED** with EFW > 215 gpm / OTSG. **If** only one OTSG is available **or** OTSG tube leak > 1 gpm, **then FEED** with EFW > 430 gpm to the good OTSG.
2. **VERIFY** an RCP is operating **or** incore temperature is stable or lowering.	**FEED** OTSG at maximum available EFW flow.
3. **VERIFY** EFW is available.	**If** SCM < 25°F, **or** [all RCPs OFF **and** incore temp is rising], **then FEED** with MFW at > 1.0 Mlbm/hr.
4. There is no minimum required flow rate.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EK1.02 (055)	
	Importance Rating	4.1	4.4

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to the Station Blackout : Natural Circulation Cooling

Proposed Question: Common 49

Plant conditions:

- Turbine/Reactor trip from 100% power.
- A loss of offsite power (LOOP) occurred.
- Both Emergency Diesel Generators tripped while starting.
- RCS pressure is 2065 psig.
- The Incore thermocouple temperature is 555 °F rising slowly.
- "A" Loop: Thot = 552 °F stable; Tcold = 495 °F lowering slowly.
- "B" Loop: Thot = 552 °F stable; Tcold = 508 °F stable.
- Both Atmospheric Dump Valves are closed.
- Emergency Feedwater Pump EF-P-1 is running.
- OTSG "A" level is at 35% and rising slowly.
- OTSG "B" level is at 35% and rising slowly.
- Both OTSGs 500 psig lowering slowly.

Which ONE of the following correctly describes the status of Natural Circulation Cooling?

A. Natural Circulation flow is verified in both loops.
B. Natural Circulation flow is NOT verified in either loop.
C. Natural Circulation flow is verified in the "A" Loop ONLY.
D. Natural Circulation flow is verified in the "B" Loop ONLY.

Proposed Answer: B. Natural Circulation flow is NOT verified in either loop.

Explanation (Optional):

A. Plausible if the examinee does not recognize Tc not tracking Tsat generator.
B. Correct – per Guide 10: ΔT is < 50 °F. & Tc not tracking OTSG Tsat.
C. Plausible if Tc lowering is thought to be indication however delta T is > 50°F.
D. Plausible if as delta T is acceptable for natural circ, however Tc not tracking OTSG Tsat.

Technical Reference(s): OP-TM-EOP-010, Guide 10, Natural Circulation (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

Guide 10
Natural Circulation

IAAT all RCPs are off, **then**:

A. **If** all of the following conditions exist, **then** adequate Natural Circulation exists:

- RCS T_{HOT} minus T_{COLD} stabilizes at less than 50°F.
- T_{HOT} < 600°F.
- Incore temperature stabilizes **and** tracks T_{HOT}.
- Cold leg temperatures approach saturation temperature for secondary side pressure.
- OTSG heat removal is indicated by feeding **or** steaming with stable OTSG pressure.
- SCM ≥ 25°F.

____ TIME Natural Circulation was VERIFIED

B. **If** OTSG tube leakage < 1 GPM and Primary to Secondary Heat Transfer exists, **then**

MAINTAIN RCS pressure above the "PREVENT RV HEAD BUBBLE" curve on Figure 1 to avoid developing a steam bubble in the Reactor Vessel head.

2. **If** one of the following conditions exists:

   - SCM < 25°F in hot leg of an isolated OTSG
   - T_{HOT} on an isolated OTSG is more than 50°F above active loop T_{HOT}.

   then FEED OTSG with EFW (or MFW if EFW is not available) until T_{HOT} A and B are within 20°F to minimize the potential for hot leg voids,
   or REDUCE RCS cooldown rate.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AK1.04 (056)	
	Importance Rating	3.1	3.2

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of saturation conditions, implication for the systems

Proposed Question: Common 50

Plant event:

- Reactor trip from 100% power due to a loss of offsite power (LOOP).
- Both Emergency Diesel Generators have failed.
- The operating crew is performing AOP-020, "LOSS OF STATION POWER", Section 4.0 – STATION BLACKOUT.
- Steam driven Emergency Feedwater Pump EF-P-1 is running.
- The Transmission System Operator reports that it appears off-site power will not be restored for 24-48 hours.
- RCS pressure is at 1815 psig.

In accordance with AOP-020, "Loss of Station Power ",which ONE of the following identifies the minimum subcooling margin (SCM) and the means of control?

A. ≥ ZERO ≤ 25 ˚F. Manually cycle the Pressurizer PORV to minimize SCM.
B. ≥ 25 ˚F. Operate Pressurizer heaters to raise RCS pressure as temperature rises.
C. ≥ 30 ˚F. Adjust MS-V-4A and MS-V-4B, as necessary to control temperature.
D. ≥ 70 ˚F. Adjust EFW flow to maintain OTSG level at 75-85% to maximize SCM.

Proposed Answer: C. ≥ 30 ˚F. Adjust MS-V-4A and MS-V-4B, as necessary to control temperature.

Explanation (Optional):

A. Plausible since some EOP's minimize subcooling and this is a means of doing so.
B. Plausible since it is a "magic number" in RULE 1 (and most EOP's) but PZR heaters are NOT available.
C. Correct – "if at any time" step 4.8 of AOP-020.
D. Plausible since it is a value used HPI throttling. The OTSG level is a RULE 4 value for when SCM is < 25 ˚F.

Technical Reference(s): OP-TM-AOP-020, Loss of Station Power Page 17 and Carryover steps (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4.8 **IAAT** SCM < 30 °F, **then ADJUST** MS-V-4A and MS-V-4B to reduce RCS temperature as necessary to maintain SCM > 30 °F.	
☐ 4.9 **IAAT** RCS pressure < 1750 psig, **then BYPASS** ESAS HPI.	
4.10 **BREAK** vacuum in the main and auxiliary condensers, **and ISOLATE** Gland Steam by performing the following:	
___ 4.10.1 **OPEN** VA-V-8 (TB 322: near west side of condenser).	
___ 4.10.2 **OPEN** VA-V-4A (TB 305: south of A aux. Condenser) **and OPEN** VA-V-4B (TB 305: south of B auxiliary Condenser).	
☐ 4.10.3 **IAAT** condenser vacuum < 10 in Hg vac, **then CLOSE** MS-V-7 (TB 322: chain operator 30' East of service building chiller).	
☐ 4.11 **IAAT** battery voltage < 105 VDC (at battery ground detector panel in inverter room) **or** an inverter trips on DC undervoltage (A-3-7, A-3-8 and A-1-6), **then PERFORM** Attachment 6A or 6B.	
4.12 **CONSERVE** station battery life by performing the following:	
___ 4.12.1 **ENSURE** all RCP DC oil lift pumps are shutdown.	
___ 4.12.2 **INITIATE** Attachment 5 to energize DCA & DCB from DC Diesel.	
☐ 4.12.3 **IAAT** Main Turbine speed < 1 rpm **and either** HG-PI-1213 < 2 psig **or** GN-P-2 is running, **then PLACE** LO-P-6 in PTL.	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	G2.4.4 (057)	
Importance Rating	4.0	4.3

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

Proposed Question: Common 51

Plant conditions:

- Reactor power 100%.

Event:

- MAP alarm A-1-6, "Inverter Failed", illuminates.
- MAP alarm A-1-8, "Battery B Discharging", illuminates.
- MAP alarm A-3-8, "Inverter 1B/1D/1F Trouble", illuminates.
- MAP alarm G-2-3, "CRD DC BKR Open", illuminates.
- 4 psig RB Pressure Channel RB3B indicates actuated.
- 500 psig ESAS Channel RC6B indicates actuated.
- 1600 psig ESAS Channel RC3B indicates actuated.
- 30 psig RB Pressure ESAS Channel RB6B indicates actuated.
- Reactor Protection Channel D has no lights illuminated.

The above conditions indicate.....

A. Loss of B DC Distribution.
B. Loss of Regulated Bus TRB.
C. Loss of 120 Volt Vital Bus D.
D. Loss of 1B 480 Volt ES MCC.

Proposed Answer: C. Loss of 120 Volt Vital Bus D.

Explanation (Optional):

A. Plausible since loss of B DC along with loss of 480V AC power to the inverter would illuminate this alarm. In addition Low Battery Voltage to the inverter would illuminate A-3-8.
B. Plausible since this is the backup power supply to the Vital Bus D and would cause these alarms if it were connected to the bus and subsequently lost.
C. Correct - alarm A-1-6 indicates loss of AC and DC to inverter. ESAS indications and RPS Channel "D" indicate loss of VBD.
D. Plausible since this would cause the A-1-6 alarm along with loss of B DC Distribution.

Technical Reference(s): OP-1107-2B, 120 Volt Vital Electrical System (Page 137) (Attach if not previously provided)
MAP Alarm A-1-6, Inverter Failed
MAP Alarm A-3-8, Inverter 1B/1D/1F Trouble

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.7
55.43 ______

Comments:

NRC QU COMMON 51

_____ 6. **CLOSE** the tie breaker SW #15 on TRA.

_____ 7. **ANNOUNCE** on plant page "Energizing Vital Bus Delta", then **CLOSE** the tie breaker, SW #30 on VBD.

_____ 8. Re-energize RPS Channel D IAW OP-TM-641-404, De-energizing RPS Channel D.

_____ 9. **VERIFY** the RC pump power Monitors status is normal by observing the lights on the front of the panels.

NOTE
ESAS Channel III is INOPERABLE if VBD is supplied by TRA

_____ 10. **If** the reactor is critical, **then MARK** this entire section N/A (i.e. DO NOT RESET the tripped ESAS channel).

_____ a. **RESET** 4 psi RB PRESS bistable by depressing the "ENABLE" pushbutton for RB3B.

_____ b. **If** RCS pressure is above bistable trip setpoint, **then PERFORM** the following:

_____ 1) **RESET** 500 psi RC PRESS bistable by depressing the "ENABLE AND CHANNEL RESET" pushbutton RC6B.

_____ 2) **RESET** 1600 psi RCS PRESS bistable by depressing the "ENABLE AND CHANNEL RESET" pushbutton RC3B.

_____ c. **If** the reactor is shutdown **and** RCS pressure is below bistable trip setpoint, **then PERFORM** the following:

_____ 1) **BYPASS** 500 psi RCS PRESS for RC6B, by pressing the "ENABLE/BYPASS" pushbutton as appropriate.

_____ 2) **BYPASS** 1600 psi RCS PRESS for RC3B, by pressing the "ENABLE/BYPASS" pushbutton as appropriate.

_____ d. **PRESS** the reset pushbutton for RB6B for 30 psi RB PRESS.

NRC QU common 51

	TMI - Unit 1 Alarm Response Procedure	Number **MAP A**
Title **Main Annunciator Panel A**		Revision No. **(See Cover Page)**

A-3-8
Revision 16

ALARM:

INVERTER 1B/1D/1F 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 < 59.5 or > 60.5 cycles per second

CAUSES:

Battery voltage high caused by battery charger trouble.
Battery voltage low caused by high battery discharge rate (580 amps for 1 hr., for example) as indicated by the ammeters in DC Dist. Panel 1B.
Overcurrent as indicated by 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
1B	1B	7
1D	1B	10
1F	1B	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 vital bus, affected RPS channel trip, and trip of 1 channel of RB isolation and cooling, HPI, and LPI in both ES actuation systems (Vital bus D affects only the "B" ES actuation system).

OBSERVATION (CONTROL ROOM):

Computer printout: Example "INVERTER 1B DC VOLTS LOW"

Inverter 1B/1D/1F DC Volts - Lo L2554/L2562/L3138
Inverter 1B/1D/1F DC Volts Hi L2555/L2563/L3137
Inverter 1B/1D/1F DC Overcurrent L2556/L2564/L3136
Inverter 1B/1D/1F Freq. Hi/Lo L2611/L2652/L3139

NRC 4th Comment 5.1

	TMI - Unit 1 Alarm Response Procedure	Number **MAP A**
Title **Main Annunciator Panel A**		Revision No. **(See Cover Page)**

A-1-6
Revision 17

<u>ALARM</u>:

INVERTER FAILED

<u>SETPOINTS</u>:

Loss of voltage at the inverter output on inverter 1A, 1B, 1C, 1D, 1E or 1F.

<u>CAUSES</u>:

1. Failure of the inverter circuitry
2. Inverter trip: breakers open→ "AC Input" <u>and</u> "Inverter Operation"("Battery Input" for 1F inverter), <u>**or**</u> "Inverter Output")
3. *Loss of both the 480V AC and 125V DC input to the inverter.*
   The AC and DC supply to each inverter are as follows:

<u>Inverter</u>	<u>AC Supply</u>	<u>DC Supply</u>	
1A	1A ES CC (Unit 1AL)	Panel 1A	SW 4
1B	1B ES CC (Unit 1AL)	1B	7
1C	1A ES CC (Unit 1BL)	1A	12
1D	1B ES CC (Unit 1BL)	1B	10
1E	1A ES CC (Unit 1CR)	1A	11
1F	1BES CC (Unit 15A)	1B	11

<u>AUTOMATIC ACTION</u>:

Loss of Vital Bus
RPS Channel trips
CRD Bkr(s) open
ES channels trip
HSPS channel trips (HSPS train power swaps to "D-16" backup if affected).
ATA or ATB xfer to backup source (if affected).
ICS inputs swap to alternate (NI Power, RCS Flow, RCS Pressure, OTSG Level).
One RCP tripped indication to all RPS channels.
Loss of various Control Room indications.

<u>OBSERVATION (CONTROL ROOM)</u>:

Affected RPS cabinet de-energized (no lights).
Tripped lights on unaffected RPS cabinets.
ES Channel trip indication on Panel PCR.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	AK1.01 (058)	
Importance Rating	2.8	3.1

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation

Proposed Question: Common 52

Plant conditions:

- Reactor shutdown is in progress due to a cell short on “A” Battery that caused failure of a lead.
- “A” Battery has been isolated.
- The A and C Battery Chargers are supplying the “A” DC Bus.

The Turbine DC Lube Oil Pump LO-P-6 was subsequently started and caused the “A” Battery Charger to go into the “current limiting” condition. The “A” Battery charger going into current limiting will __________.

A. result in a voltage dip on the DC Bus
B. cause the A Battery Charger to trip on overcurrent
C. cause A Battery Charger to shift to the equalize mode
D. prevent the C Battery Charger from going into the current limiting condition

Proposed Answer: A. result in a voltage dip on the DC Bus

Explanation (Optional):

A. Correct answer based on a CAUTION in 1202-9A, Loss of A DC Distribution System. With the battery isolated, there is no large capacity to draw on.
B. Plausible because this is an excessive current condition but will not trip the charger.
C. Plausible since equalize mode raises voltage but the automatic shift will not occur.
D. Plausible if the examinee thinks this condition will limit the current on the bus; however it has no preventive effect on the C Battery Charger.

Technical Reference(s): EP 1209-9A, Loss of A DC Distribution System (Page 8 CAUTION) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: __________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

CAUTION
Unless connected to the Station Battery, the battery chargers may go into "current limiting" and cause a DC voltage dip when large DC motors are started.

16. **IF** battery chargers can be restored **AND** the "A" station battery is **NOT** available, **THEN PLACE** the following motors in pull-to-lock:

_____ a. RC-P-2A-2

_____ b. RC-P-2C-2

_____ c. LO-P-9A

_____ d. LO-P-6

17. **IF** restoring power to substation panel "DCA" **AND** 1E DC Distribution Panel is **NOT** available, **THEN PERFORM** the following:

_____ a. **OPEN** Switch #12 on "DCA" (1E ES Dist. Pnl. Feed)

_____ b. **INSTALL** 3 fuses in SW #11 cubicle on "DCA" (Fuses Stored in SF Desk).

_____ c. **CLOSE** Switch #11 on "DCA" (DC diesel and DCB Cross Tie)

_____ d. **CLOSE** Switch #1 on "DC Diesel"

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	AK3.03 (062)	
Importance Rating	3.6	3.9

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS

Proposed Question: Common 53

Nuclear Services Closed Cooling Water (NSCCW) Containment Isolation Valves NS-V-4, NS-V-15 and NS-V-35 close on a Line Break Isolation actuation to prevent......

A. dilution of the Reactor Building Sump water due to a NSCCW system leak inside containment.
B. a radioactive release path in the event of a NSCCW break and an RCS break in containment.
C. loss of bearing cooling water to the Reactor Building Cooling Fans AH-E1A/1B/1C during an emergency event.
D. loss of cooling water flow to essential components outside the RB in the event of a NSCCW leak inside containment.

Proposed Answer: B. a radioactive release path in the event of a NSCCW break and an RCS break in containment.

Explanation (Optional):
A. Plausible since a leak inside the containment would put non-borated water into the RB Sump.
B. Correct - NS-V-4, 15, 35 all close on Line Break Isolation. The reason is described in the ESAS lesson plan.
C. Plausible since NSCCW does supply bearing cooling to the AH-E-1s; however they are supplied through a separate line to the RB.
D. Plausible since the NSCCW System does continue to supply components outside the containment during an emergency event.

Technical Reference(s): TQ-TM-104-642-C001, Engineered Safeguards Actuation System (Page 36,37) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: 642-GLO-5 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

Content/Skills	Activities/Notes
6) Alarm DH-V-1/2 Auto Close Defeated – PLA-7-8. a) Alarm conditions (1) Both 400 psig bistables tripped AND (2) Either DH-V-1 OR 2 Key Switch in bypass position. b) Used when the Reactor Head is removed and feature is intentionally bypassed to prevent unintentional closure and loss of Decay Heat Removal. (1) By procedure, the breakers to DH-V-1 and DH-V-2 are opened to prevent unintentional closure 7) Emergency Condition a) While operating on DH, if some transient were to raise Reactor Coolant System pressure (RC3A/RC3B-P13) to ≥400 psig, DH-V-1 and/or DH-V-2 will auto close and MAP Alarm C-1-6 will annunciate. b) If DH-V-1 or 2 auto closes while operating in Decay Heat Mode, the operator must secure the operating Decay Heat Removal Pump until an adequate suction source can be established c) Refer to Loss of Decay Heat Removal Emergency Procedures. h. Line Break Isolation 1) Line Break Actuation Description: a) Line Break Isolation is energized to actuate and closes valves in the Intermediate Closed Cooling Water System and the Nuclear Service Closed Cooling Water System. b) The purpose of the valve closures is to isolate possible pathways from the RB to the outside environment.	**Using PLA-7-8 to describe the DH-V-1/2 Auto Close Defeated alarm response.**

Content/Skills	Activities/Notes
2) Line Break Isolation Actuation Conditions	
a) ES actuation signal (1600 psig, 500 psig Reactor Coolant System pressure or 4 psig Reactor Building pressure)	
AND	
b) Low Level in Intermediate Closed Cooling Water surge tank and/or Nuclear Services Closed Cooling Water surge tank	
3) Affected components:	
a) NS-V-4, 15, 35	
b) IC-V-2, 3, 4, 6	
4) Clearing/resetting Actuation signal	
a) Clear Engineered Safeguards Actuation signal	
OR	
b) Refill Surge Tank and reset both channels.	
5) Indications	**Use Jump to alternate between slides for indication.**
a) Console	
b) PCR	
6) **(RO)** Electrical Print Description:	**Electrical Print 209-542**
a) Low Nuclear Services Closed Cooling Water or Intermediate Closed Cooling Water Surge Tank level and High Pressure Injection	
b) Nuclear Services Closed Cooling Water Surge Tank level low shuts a contact energizing the (63X1) relay.	
c) (63X1) contacts shut.	
d) If High Pressure Injection is actuated contact (74X3/RC) shuts, energizing the (63Z1) relay.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EK2.2 (E04)	
	Importance Rating	4.2	4.2

(K&A Statement) Knowledge of the interrelations between the (Inadequate Heat Transfer) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: Common 54

In EOP-002, "LOSS OF 25 °F SUBCOOLED MARGIN", Step 4.1 reads "**IAAT** HPI flow is ADEQUATE, **then GO TO** Step 3.9".

Which one of the following meets the facility criteria for "HPI flow is ADEQUATE"?

A. Pressurizer level is rising.
B. HPI flow ≥ 200 gpm is verified.
C. In-core temperatures have stabilized.
D. All components in HPI Train "A" are in their ES actuation alignment.

Proposed Answer: D. All components in HPI Train "A" are in their ES actuation alignment.

Explanation (Optional):

A. Plausible since, under normal but not all circumstances, this would indicate increasing inventory.
B. Plausible since this is close to the criteria (OP-TM-211-901, Attachment 7.4) for 2400 psig but well below all other conditions.
C. Plausible since this is an EOP criteria for determining if heat removal capability is available.
D. Correct – meets OS-24 (Conduct of Operations During Abnormal and Emergency Events) definition 3.2.

Technical Reference(s): OS-24, Page 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

1.0 **PURPOSE**

Establish the structure, responsibilities, and guidance to promote consistent and reliable operator performance when responding to emergency events which require implementing usage level "EP" Procedures.

2.0 **APPLICABILITY/SCOPE**

Operations Department personnel performing or responding to direction provided in "EP" Usage level procedures. This procedure only applies when an EVENT procedure is entered.

3.0 **DEFINITIONS**

3.1 ADEQUATE CORE COOLING:

A demonstrated cooldown rate accomplished by removing core decay heat either through primary to secondary heat transfer, RCS feed and bleed or a combination of both methods. A core cooldown rate of 40°F/hr demonstrates core heat removal. At less than 255°F, adequate core cooling may be demonstrated by stable or decreasing in core temperature.

3.2 ADEQUATE HPI

HPI flow is "adequate" when flow exceeds the flow assumed in the ECCS analysis. Where the EOP calls for "adequate HPI", this condition may be confirmed by (1) all components of one HPI train are in their ES condition, or (2) Indicated HPI flow exceeds flow on Attachment 7.4 in OP-TM-211-901.

3.3 APPROACHING:

The parameter is trending toward the setpoint or limit, and based on its trend and plant conditions it is likely that the parameter will reach the setpoint or limit.

3.4 CARRYOVER STEP:

A conditional step which uses the IF AT ANY TIME (IAAT) or WHEN AT ANY TIME (WAAT) logic. A Carryover step applies after that step is reached and throughout the remainder of the procedure or until it is no longer listed on the CARRYOVER PAGE. When the procedure is complete, the step does not apply. As an aid to the procedure user, CARRYOVER STEP(s) may appear on the facing page following the appearance of the original step and remain in the individual procedure until no longer applicable. (see Section 4.1.11 for usage details)

3.5 ENSURE:

If the system / component is not already in the desired condition, then take action to place the system / component in the specified condition. e.g. ENSURE MU-V-18 is closed. If the performer finds the valve open, the performer is to close the valve. The verb does authorize local operation of the valve, but does not authorize operation of other components to achieve the intent of the procedure.

Exception: ENSURE and VERIFY in Emergency (1202 series) and Abnormal (1203 series) procedures are used interchangeably until they are revised.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EK3.2 (E05)	
	Importance Rating	3.5	4.0

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the (Excessive Heat Transfer): Normal, abnormal and emergency operating procedures associated with (Excessive Heat Transfer).

Proposed Question: Common 55

Plant conditions:

- Reactor tripped due to steam leak from OTSG 1A in the Reactor Building
- 1600 psig and 4 psig ESAS actuations occurred and have been bypassed
- RCS pressure 1620 psig
- RCS Tave 522°F, steady
- Pressurizer level 80 inches and rising
- RB pressure 4.2 psig
- Rule 3 isolation OTSG 1A has been completed
- Rule 2, HPI/LPI Throttling is in progress

Given the above conditions, which ONE of the following would be the reason for leaving MU-V-14A and MU-V-14B open when performing OP-TM-211-901, "Emergency Injection (HPI/LPI)", Attachment 7.3, Throttling HPI?

A. To maintain adequate shutdown margin with RCS temperature < 525°F.
B. To maintain minimum Makeup Pump flow requirements until HPI is secured.
C. To maintain adequate suction to the Makeup Pump until Pressurizer level recovers to level setpoint.
D. To raise RCS boron concentration in anticipation of the subsequent cooldown to cold shutdown conditions.

Proposed Answer: B. To maintain adequate shutdown margin with RCS temperature <525°F.

Explanation (Optional):

A. Correct answer. OP-TM-211-901 Attachment 7.3 - Emergency Boration is required IAW EOP-003.
B. Plausible since MU-V-36 and MU-V-37 must be opened for flow prior to reducing HPI flow to maintain minimum flow requirements.
C. Plausible since Pressurizer level is less than setpoint.
D. Plausible since RCS cooldown will be required to repair the OTSG.

Technical Reference(s): OP-TM-211-901, Attachment 7.3 Throttling HPI (Attach if not previously provided)

EOP-003, Excessive Primary to Secondary Heat Transfer (Page 3)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

Question Source: Bank # ________

Modified Bank # ________ (Note changes or attach parent)

New X

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.10

55.43 ________

Comments:

NRC QU comment 5-5

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 **ENSURE** RCS temperature reduction has been terminated.	___ **If** PSHT is **not** excessive **and** temperature reduction is due to HPI/Break Cooling, **then GO TO** EOP-006
___ 3.3 **VERIFY** primary to secondary heat transfer is being established.	___ **GO TO** EOP-004
___ 3.4 **VERIFY** RCS Tcold > 525 °F.	___ **INITIATE** Emergency boration – Rule 5, EB.
___ 3.5 **ENSURE** performance of an alarm review.	
___ 3.6 **REQUEST** SM evaluate Emergency Action Levels (EALs).	
___ 3.7 **VERIFY** OTSG B is providing sufficient steam for Gland Steam.	___ 1. **ENSURE** Aux Boiler is operating. 2. **TRANSFER** gland sealing steam to the Auxiliary Steam supply as follows: ___ A. **OPEN** AS-V-8 (TB 355' South of 6th stage drain collection tank). ___ B. **CLOSE** GS-V-4

NRC QU Common 55

ATTACHMENT 7.3
THROTTLING HPI
Page 1 of 1

1 **VERIFY** ESAS in defeat IAW OP-TM-642-901, "1600 psig ESAS Actuation" ___

2 **IAAT** three MU pumps are running **and** CRS concurrence is obtained, **then SHUTDOWN** the ES selected pump lined up to MU & SI **and PLACE** CS in Normal-After-Stop. (e.g. normally MU-P-1A) ☐

3 **VERIFY** throttling is permitted IAW RULE 2 **and OBTAIN** CRS concurrence ___

4 **WAAT** HPI throttling is permitted IAW RULE 2 **and** prior to reducing any MU pump flow to less than 115 GPM, **then** perform the following: ☐

1. **If** DH-V-7A **and** DH-V-7B are Closed, **then OPEN** MU-V-36 **and** MU-V-37 ___
2. **If** DH-V-7A **or** DH-V-7B are Open, **then** ___
   1) **OPEN** RC-V-2 **and** RC-RV-2
   2) **ENSURE** any MU pump opposite MU & SI is shutdown

5 **WAAT** Emergency Boration is **<u>not</u>** required (Rule 5), **then INITIATE** Guide 9 to close MU-V-14A and MU-V-14B ☐

6 **If** CRS directs "termination" of HPI, **then** ___
   A. **SHUTDOWN the** MU pumps which started on ES **and PLACE** CS in Normal-after-stop
   B. **CLOSE** both MU-V-16 valves lined up to MU/SI pump
   C. **CLOSE** both MU-V-16 valves opposite MU/SI
   D. **GO TO** Step 10

7 **CLOSE** MU-V-16 valves to establish flow through one valve on each train and retain HPI flow through four RCS nozzles (e.g. MU-V-16A and MU-V-16D) ___

8 **THROTTLE** the open MU-V-16 opposite of MU and SI. ___

9 **If** two MU pumps are operating, **then when** flow is reduced to one MU-V-16, ___

1. **If** <u>both</u> pumps operating are ES selected, **then SHUTDOWN** pump opposite MU & SI **and PLACE** CS in Normal-After-Stop. (e.g. normally NA) ___
2. **If** two MU pumps are operating, **then SHUTDOWN** the ES selected pump **and PLACE** CS in Normal-After-Stop. (e.g. normally MU-P-1C) ___

10 **When** OP-TM-244-901 criteria is satisfied, **then OPEN** MU-V-18 ___

11 **If** MU-V-36 or MU-V-37 is Closed, **then ENSURE** MU or SI flow > 40 GPM. ___

12 **THROTTLE** MU-V-16 parallel to MU and SI (i.e. normally MU-V-16B) ___

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EK3.4 (E10)	
	Importance Rating	4.0	4.0

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the (Post-Trip Stabilization): RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Proposed Question: Common 56

Plant conditions:

- Reactor trip from 100% power caused by a turbine trip.
- All control rods are inserted.
- All immediate actions are complete.

Which ONE of the following requires an immediate transition out of EOP-001, "REACTOR TRIP"?

A. Loss of ICS AUTO Power.
B. Tave has stabilized at 535 °F.
C. In-core temperatures 560 °F rising with only Emergency Feedwater Pump EF-P-1 running.
D. RM-A-5, CONDENSER VACUUM PUMP EXHAUST, in ALARM with Pressurizer level lowering at 1 inch per minute.

Proposed Answer: D. RM-A-5, CONDENSER VACUUM PUMP EXHAUST, in ALARM with Pressurizer level lowering at 1 inch per minute.

Explanation (Optional):

A. Plausible since this requires EOP-001 action but not transition.
B. Plausible since it is one part of the criteria for excessive cooldown.
C. Plausible since this meets half the criteria for inadequate HT.
D. Correct. Clearly indicates OTSG tube leakage > 1 gpm.

Technical Reference(s): EOP-001, Step 3.1 (Attach if not previously provided)
OS-24 definitions

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ______

	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	B.10	
	55.43		

Comments:

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** EOP-002 2. XHT **GO TO** EOP-003 3. LOHT **GO TO** EOP-004 4. OTSG tube leakage > 1 gpm **GO TO** 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** OTSG A **and** B Operating Range level < 97.5%	___ **TRIP** both MFW pumps.
___ 3.5 **VERIFY** Main FW Flow to A **and** B OTSG are each < 0.5 mlb/hr.	___ **ENSURE** FW-V-5A **and** FW-V-5B are CLOSED.
___ 3.6 **VERIFY** OTSG level > setpoint.	___ **INITIATE** RULE 4 – FWC.
___ 3.7 **VERIFY** ICS/NNI HAND **or** AUTO Power is available.	___ **INITIATE** OP-TM-AOP-025 "Loss of ICS/NNI Hand and Auto Power."

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	AK3.02 (037)	
Importance Rating	3.2	3.5

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Comparison of makeup flow and letdown flow for various modes of operation

Proposed Question: Common 57

Plant conditions:

- The reactor was manually tripped
- One rod in Group 3 failed to insert into the core.
- OP-TM-EOP-005, "OTSG TUBE LEAKAGE", was entered due to identified tube leakage.
- RCS Tave is 528 °F and lowering slowly.
- RCS Pressure is 1920 psig and lowering slowly.
- Pressurizer level is 100 inches and stable.
- Makeup Tank level is 88 inches.
- Makeup Tank pressure is 25 psig.
- Makeup flow is 8 gpm.
- Seal Injection flow is 32 gpm.
- Letdown flow is 45 gpm.
- MU-V-14A is open.

Which ONE of the following actions is required?

A. Adjust RCP seal injection flow to establish 42 gpm.
B. Initiate RCS bleed to RC Bleed Tank to lower makeup tank level.
C. Open MU-V-14B to ensure adequate NPSH to the Makeup Pump.
D. Raise letdown flow to 60 gpm to increase Emergency Boration Flow.

Proposed Answer: D. Raise letdown flow to 60 gpm to increase Emergency Boration Flow.

Explanation (Optional):

A. Plausible since seal injection is lower than normal but meets requirements. Incorrect band allows 32-40.
B. Plausible if makeup tank level and pressure inside allowable band are not recognized.
C. Plausible since Emergency Boration is in progress but Rule 5 requires opening only one MU-V-14 and the Makeup Tank is still available.
D. Correct to achieve Emergency Boration 50 gals per EOP Rule 5 step 4.

Technical Reference(s):	OP-TM-EOP-010, Rule 5, Emergency Boration	(Attach if not previously provided)
	OP-TM-EOP-005, OTSG Tube Leakage (Page 3)	

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 3.7 **NOTIFY** SM to evaluate Emergency Action Levels (EALs).	
3.8 **IDENTIFY** the affected OTSG. ____ **OTSG A** ____ **OTSG B**	____ **REQUEST** OTSG Steam line samples. ____ **REQUEST** Steam line Radiation Surveys. ____ **OBSERVE** OTSG levels and feed rates.
____ 3.9 **INITIATE** Attachment 2.	
☐ 3.10 **IAAT** Aux. Steam is available, **then** ____ 1. **OPEN** AS-V-8 (TB 355' south of 6th stage drain collection tank). ____ 2. **CLOSE** GS-V-4. ____ 3. **TRANSFER** operating FWPs to Aux Steam: ____ **SLOWLY OPEN** AS-V-5A (7' W of FW-P-1A 2' up). ____ **SLOWLY OPEN** AS-V-5B (7' W of FW-P-1B 2' up).	
____ 3.11 **VERIFY** the reactor is critical.	____ **GO TO** Step 3.17.
____ TIME	
____ 3.12 **When** RCS Tavg < 555°F, **then CONTINUE**.	
____ 3.13 **PLACE** TBVs to HAND.	
____ 3.14 **TRIP** the reactor **and THROTTLE** TBVs to prevent MSSV lift and stabilize RCS temperature.	
____ 3.15 **VERIFY** Groups 1 through 7 have fully inserted.	____ **INITIATE** Rule 5, "Emergency Boration."
____ 3.16 **PERFORM** a SYMPTOM CHECK.	
____ 3.17 **ENSURE** announcement of reactor trip over the plant page and radio.	
____ 3.18 **DISPATCH** an Operator to check MSSV status.	
3.19 **ENSURE** performance of an alarm review.	

5

EB

Rule 5
Emergency Boration

IAAT any of 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** lowering as expected

then Emergency Borate as follows:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **WAAT** one of the following conditions exist: - 1% dk/k SHUTDOWN has been achieved for the expected plant condition IAW Figure 1 of 1103-4, "Soluble Poison Concentration Control", **or** 1103-15A, "SDM and Reactivity Balance" - LPI > 1250 GPM per line - Tavg > 525°F and stable or rising **and** all rods are inserted **and** Neutron flux is lowering as expected **then** emergency boration **may** be terminated.	
2. **VERIFY** a MU pump is operating.	**INITIATE** OP-TM-AOP-041 "Loss of Seal Injection."
3. Perform one of the following: ____ **OPEN** MU-V-14A. ____ **OPEN** MU-V-14B. ____ **PERFORM** Guide 1 "Emergency Boration Backup Methods."	
4. **VERIFY** Total Injection (MU, SI & HPI) > 50 GPM.	1. **INITIATE** OP-TM-211-950, "Restoration of Letdown Flow". 2. **INITIATE** OP-TM-211-441, "Increased Letdown Flowrates". 3. **If** MU tank level > 92" or MU tank pressure > 34 psig, **then INITIATE** OP-TM-211-462, "Lowering RCS/MU Volume Bleed".
5. **STOP** any activities which may be diluting RCS boron concentration.	
6. **If** SCM > 25°F and neutron flux indication is rising, **then STABILIZE** RCS temperature.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	AA1.04 (051)	
	Importance Rating	2.5	2.5

(K&A Statement) Ability to operate and / or monitor the following as they apply to the Loss of Condenser Vacuum: Rod position

Proposed Question: Common 58

Plant conditions:

- A plant startup is in progress in accordance with 1102-2, "Plant Startup".
- Reactor power is 45%.
- Tave is 579 °F.
- Turbine load is 385 MWe.

Event:

- MAP Alarm N-1-6, "MN COND VACUUM LO actuates".
- Condenser vacuum is degrading slowly.

Assuming no operator action, which ONE of the following describes the relative position of the control rods as compared to pre-degrading vacuum position?

A. LOWER due to a reduction in turbine efficiency.
B. LOWER due to a rise in turbine efficiency.
C. HIGHER due to a reduction in turbine efficiency.
D. HIGHER due a rise in turbine efficiency.

Proposed Answer: C. HIGHER due to a reduction in turbine efficiency.

Explanation (Optional):

A. Plausible because the reason is correct but rod direction is not.
B. Plausible if examinee misunderstands ICS/ULD response.
C. Correct answer. With the ICS in AUTO, as backpressure rises plant efficiency is reduced and control rods will withdraw to maintain ULD demanded load.
D. Plausible since control rods will withdraw; however condenser backpressure will rise and turbine load will reduce.

Technical Reference(s): OP-TM-AOP-0101 Pgs 5&6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	____________	(As available)	
Question Source:	Bank #	____	
	Modified Bank #	____	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	____	
Question Cognitive Level:	Memory or Fundamental Knowledge	____	
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.5	
	55.43	____	

Comments:

5.0 FOLLOW UP ACTIONS

Step 3.1 Entry into any EOP or AOP is announced to ensure all operators are aware of major changes in plant conditions. This action also makes chemistry, maintenance and radiation protection personnel aware.

Step 3.2 Whenever the 1A 4160V bus is energized, the performer is directed to the return to normal section.

Step 3.3 The reduction in CW flow will cause secondary efficiency to degrade. If the ULD is in HAND, the ICS will withdraw control rods to attempt to maintain target load. The ULD demand must be reduced to prevent rod withdraw which could result in reactor power above 2568 MWt.

Step 3.4 A reactor power reduction will be required if a FW pump trips or if the degraded CW flow would result in excessive turbine backpressure at the current turbine load. Either of these conditions could be the limiting case depending on ambient conditions. A "forced" power reduction is performed IAW 1102-4. The critical actions identified in 1102-4 are performed during the load reduction, the remaining actions are deferred until plant conditions are stable.

Step 3.5 CW Flow is effected by two consequences of the loss of 1A 4160V bus. (1) CW-P-1A & CW-P-1D are shutdown and (2) CW-V-1A and CW-V-1D lose power, and fail as is. The flow from the remaining CW pumps is significantly reduced by reverse flow through CW-P-1A & CW-P-1D. Prompt action to close these valves can raise CW flow significantly. There is a potential for the second FW pump turbine to trip on low vacuum depending on ambient conditions, air in leakage, CW system resistance (i.e. debris in aux condenser). (reference 7.1)

Step 3.6 Generator bus duct cooling does not auto transfer to the redundant train. If GN-E-1A was in service, local action to swap dampers is required. The urgency for this action is dependent upon plant load. If a runback was not required (e.g. cold CW and abnormal CO lineup), then bus duct cooling needs prompt action (i.e. restore bus duct cooling or reduce load within 30 minutes).

To facilitate prompt response OP-TM-715-451 is posted at the bus duct coolers.

Step 3.7 If EHC-P-1A was operating, then it will shutdown. EHC-P-1B should start automatically on low EHC pressure. This step is to verify this has occurred or start EHC-P-1B. This step is early in the procedure sequence to avoid potential turbine trip.

Step 3.8 If GN-P-5A was operating, then it will shutdown. GN-P-5B should start automatically on low GSC flow. This step is to verify this has occurred or start GN-P-5B. This step is early in the procedure sequence to avoid potential turbine trip.

Step 3.9 VA-P-1A and VA-P-1C will trip if operating. VA-P-1B should start automatically on CS mismatch. This step is to verify this has occurred or start VA-P-1B. This step is early in the procedure sequence to prevent further degradation of condenser vacuum or turbine trip.

Step 3.10 VA-P-2A and VA-P-2C will trip if operating. VA-P-2B should start automatically on CS mismatch. This step is to verify this has occurred or start VA-P-2B. This step is early in the procedure sequence to prevent further degradation of condenser vacuum. NOTE: Prompt action to close CW-V-1A & CW-V-1D is the best action to protect the FWPT from low vacuum trip.

Step 3.11 SC-P-1A will trip if operating. SC-P-1B or SC-P-1C should start automatically on CS mismatch. Two SC pumps should be operating. This step is to verify this has occurred or start SC-P-1B or SC-P-1C. This step is early in the procedure sequence to avoid high generator temperatures or other conditions caused by inadequate secondary closed cooling.

NOTE The NOTE separates the most time critical actions of the procedure from the remainder of the procedure.

Two attachments are included in the procedure to supplement operator understanding of the event and guide actions when events occur beyond those covered in the procedures.

Attachment 1 is a summary of the significant effects of the loss of the bus. "Significant" effects were those which impact personnel safety, nuclear safety or power production. Degradation of redundancy of critical equipment was identified. Any actions required by Tech Spec, ODCM or AP 1038 were identified. Loss of equipment which would not have an affect within 4 hours was not identified as "significant". The evaluation was performed using the 100% power normal plant alignment with no out of service equipment. The effect on plant reactor trip response was also considered.

Attachment 2 is a simplified electrical diagram showing all electrical distribution centers fed from the 1A 4160V bus.

Step 3.12 A FW pump trip is expected. After reactor power is stable this step is performed to ensure standby pumps are operating if available. CO-P-2C cannot be started until power to CO-P-9C is restored.

If the event occurred at lower reactor power or with reactor shutdown, then the number of pumps required at the current plant condition per 1102-4 should be operating.

Step 3.13 This step is to ensure an alarm review has been performed. This is standard practice following any transient, as described in OS-24, and may have been completed earlier if resources were available.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	AA1.02 (060)	
Importance Rating	2.9	3.1

(K&A Statement) Ability to operate and / or monitor the following as they apply to the Accidental Gaseous Radwaste: Ventilation system

Proposed Question: Common 59

Plant conditions:

- The plant is at 100% power.
- Waste Gas Decay Tank 1B is in service.
- Waste Gas Decay Tank 1A is at 88 psig and isolated.
- Waste Gas Decay Tank 1C is at 0 psig and in standby.

Event:

- Waste Gas Decay Tank 1A relief valve has failed open.

Which ONE of the following radiation monitors will respond to the rise in activity?

A. RM-A-4, FUEL HANDLING BUILDING VENTILATION DUCT MONITOR.
B. RM-A-6, AUXILIARY BUILDING VENT EXHAUST.
C. RM-A-7, GAS WASTE TANK DISCHARGE MONITOR.
D. RM-A-8, AUX AND F.H.B. EXH DUCT.

Proposed Answer: D. RM-A-8, AUX AND F.H.B. EXH DUCT.

Explanation (Optional):

A. Plausible since it monitors a ventilation flow stream.
B. Plausible since it monitors a ventilation flow stream.
C. Plausible since it monitors a flow stream associated with the WDG-T.
D. Correct answer. Relief discharges to a duct upstream of RM-A-8.

Technical Reference(s): Drawing 302841 (Attach if not previously provided)
Drawing 302694

Proposed references to be provided to applicants during examination: None

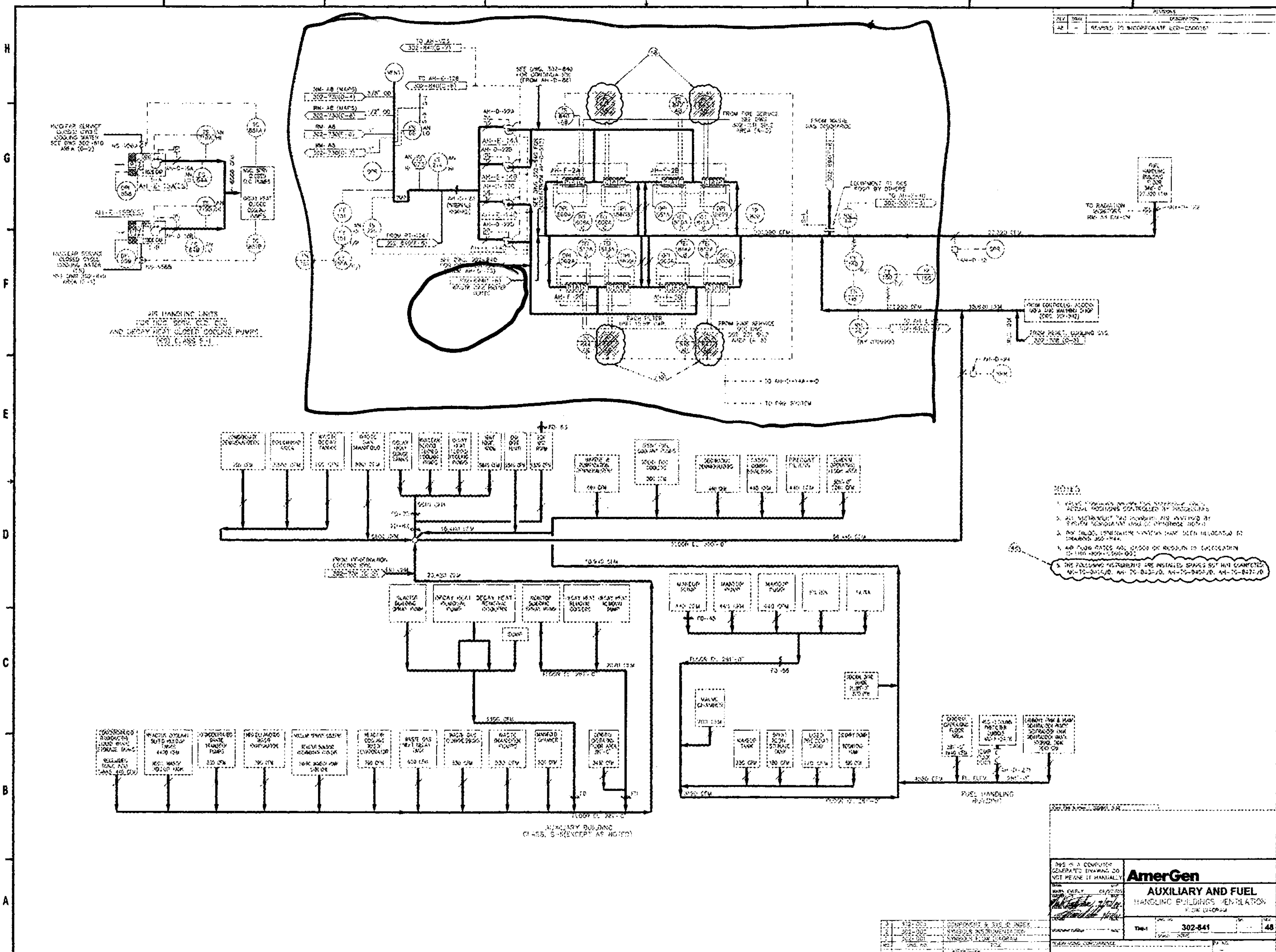
Learning Objective: ______ (As available)

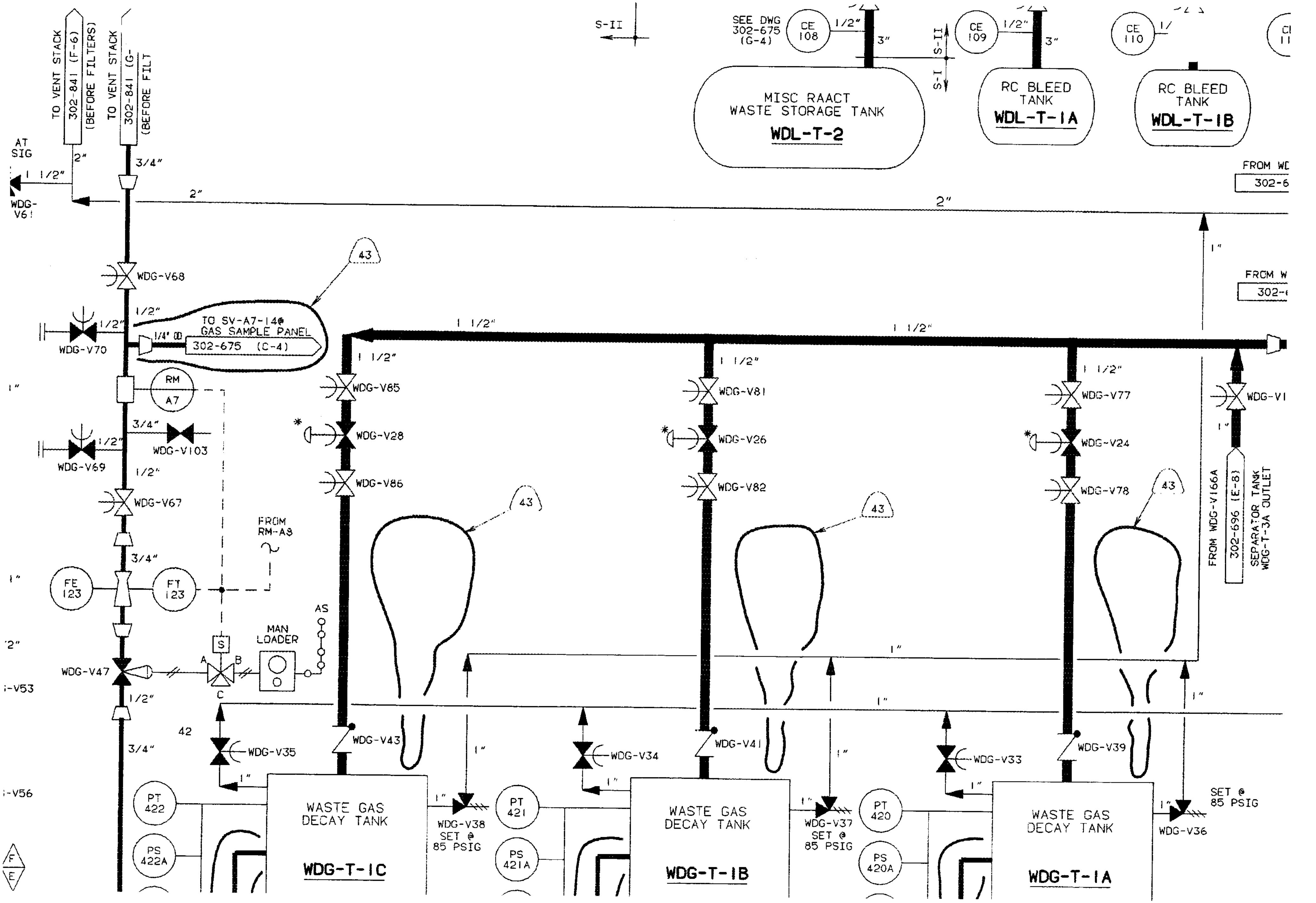
Question Source: Bank # ______

	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.11	
	55.43		

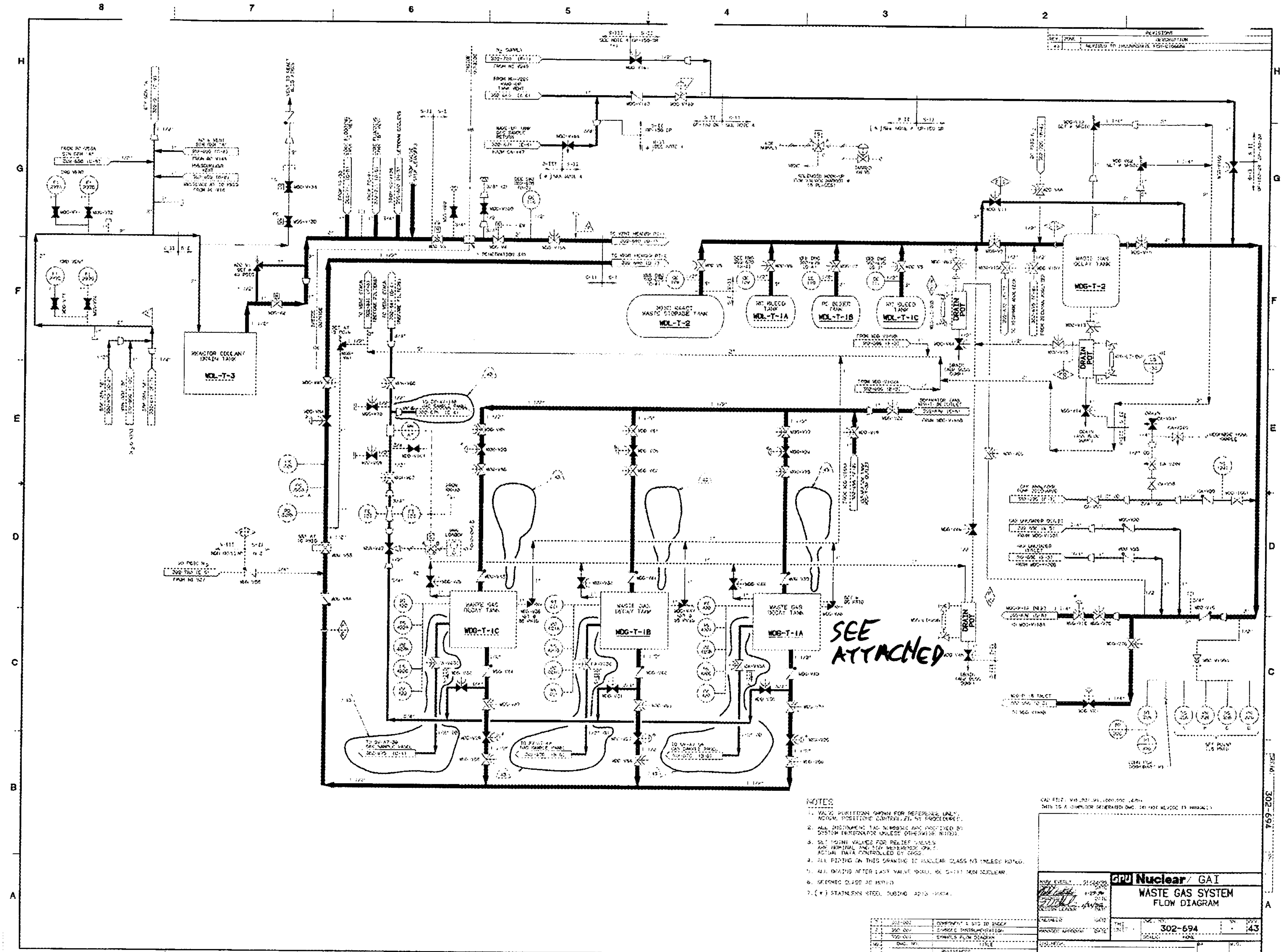
Comments:

SI - ATTACHED

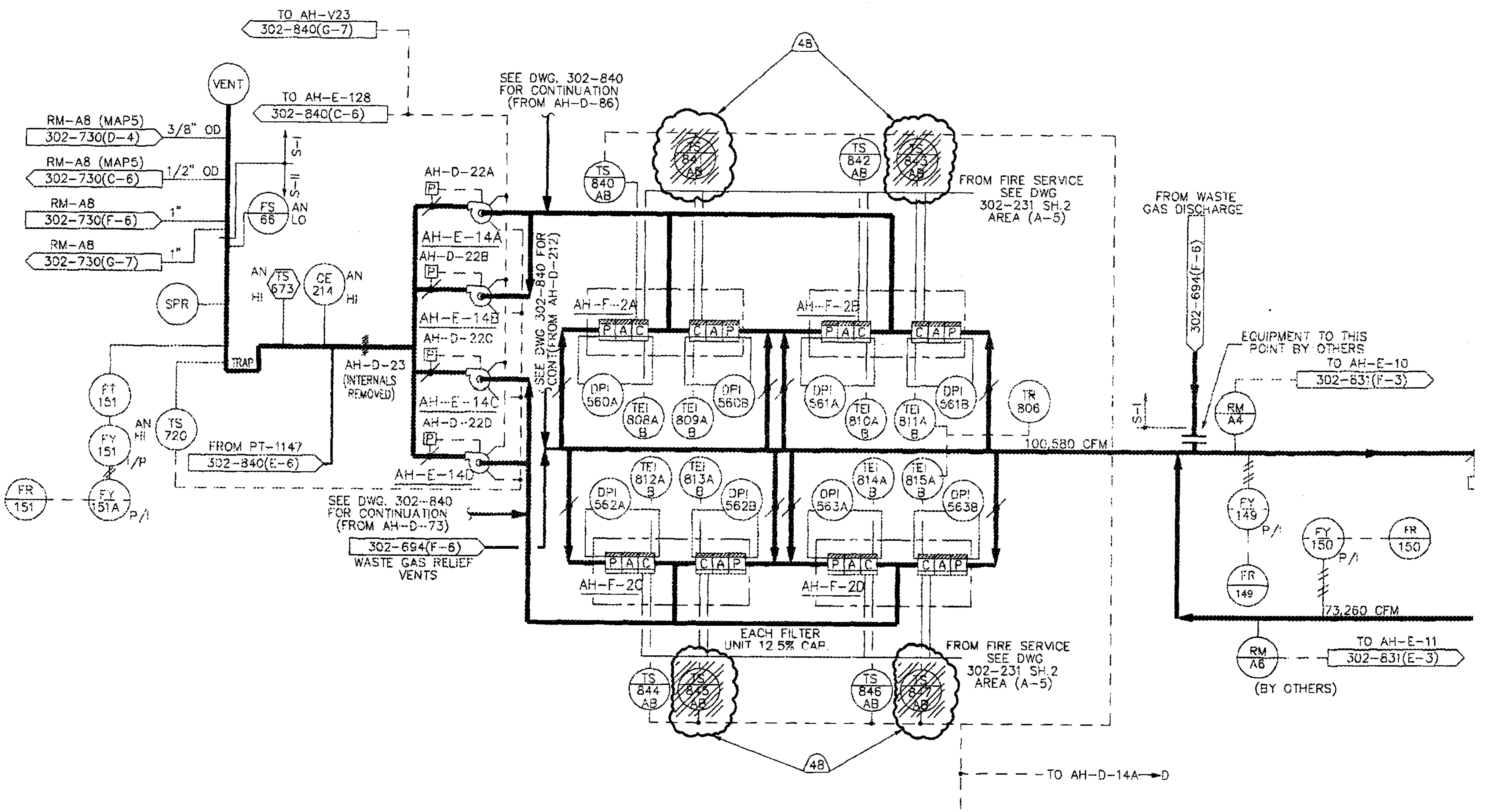

AmerGen
AUXILIARY AND FUEL
HANDLING BUILDINGS VENTILATION
FLOW DIAGRAM
302-841
AIR HANDLING UNITS
AUXILIARY BUILDING
FUEL HANDLING BUILDING
NOTES



TO VENT STACK 302-841 (F-6) (BEFORE FILTERS)
TO VENT STACK 302-841 (G- (BEFORE FILT
S-II
SEE DWG 302-675 (G-4)
CE 108
CE 109
CE 110
MISC RAACT WASTE STORAGE TANK
WDL-T-2
RC BLEED TANK
WDL-T-1A
RC BLEED TANK
WDL-T-1B
S-I
FROM WD
WDG-V68
WDG-V70
TO SV-A7-14@ GAS SAMPLE PANEL
302-675 (C-4)
RM A7
WDG-V103
WDG-V69
WDG-V67
FROM RM-A8
FE 123
FT 123
MAN LOADER
AS
WDG-V47
WDG-V35
WDG-V65
WDG-V28
WDG-V86
WDG-V43
WDG-V81
WDG-V26
WDG-V82
WDG-V41
WDG-V34
WDG-V77
WDG-V24
WDG-V78
WDG-V39
WDG-V33
FROM WDG-V166A
302-696 (E-8)
SEPARATOR TANK WDG-T-3A OUTLET
PT 422
PS 422A
WASTE GAS DECAY TANK
WDG-T-1C
WDG-V38 SET @ 85 PSIG
PT 421
PS 421A
WASTE GAS DECAY TANK
WDG-T-1B
WDG-V37 SET @ 85 PSIG
PT 420
PS 420A
WASTE GAS DECAY TANK
WDG-T-1A
SET @ 85 PSIG
WDG-V36



REACTOR COOLANT DRAIN TANK
WDL-T-3
MISC. WASTE STORAGE TANK
WDL-T-2
RC BLEED TANK
WDL-T-1A
WDL-T-1B
WDL-T-1C
WASTE GAS DELAY TANK
WDG-T-2
DRAIN POT
WASTE GAS DECAY TANK
WDG-T-1C
WDG-T-1B
WDG-T-1A
SEE ATTACHED
NOTES
GPU Nuclear / GAI
WASTE GAS SYSTEM
FLOW DIAGRAM
302-694
43



TO AH-V23
302-840(G-7)
VENT
TO AH-E-128
302-840(C-6)
RM-A8 (MAP5)
302-730(D-4)
3/8" OD
RM-A8 (MAP5)
302-730(C-6)
1/2" OD
RM-A8
302-730(F-6)
1"
RM-A8
302-730(G-7)
1"
FS 66 AN LO
TS 673 AN HI
CE 214 AN HI
SPR
TRAP
FT 151
FY 151 I/P
FY 151A P/I
FR 151
TS 720 AN HI
FROM PT-1147
302-840(E-6)
AH-D-23
(INTERNALS REMOVED)
SEE DWG. 302-840 FOR CONTINUATION (FROM AH-D-86)
SEE DWG 302-840 FOR CONT (FROM AH-D-212)
SEE DWG. 302-840 FOR CONTINUATION (FROM AH-D-73)
302-694(F-6)
WASTE GAS RELIEF VENTS
AH-D-22A
AH-E-14A
AH-D-22B
AH-E-14B
AH-D-22C
AH-E-14C
AH-D-22D
AH-E-14D
AH-F-2A
AH-F-2B
AH-F-2C
AH-F-2D
DPI 560A
DPI 560B
DPI 561A
DPI 561B
DPI 562A
DPI 562B
DPI 563A
DPI 563B
TEI 808A B
TEI 809A B
TEI 810A B
TEI 811A B
TEI 812A B
TEI 813A B
TEI 814A B
TEI 815A B
TS 840 AB
TS 841 AB
TS 842 AB
TS 843 AB
TS 844 AB
TS 845 AB
TS 846 AB
TS 847 AB
48
FROM FIRE SERVICE SEE DWG 302-231 SH.2 AREA (A-5)
EACH FILTER UNIT 12.5% CAP.
TR 806
100,580 CFM
FROM WASTE GAS DISCHARGE
302-694(F-6)
EQUIPMENT TO THIS POINT BY OTHERS
TO AH-E-10
302-831(F-3)
RM A4
FY 149 P/I
FR 149
FY 150 P/I
FR 150
73,260 CFM
RM A6
(BY OTHERS)
TO AH-E-11
302-831(E-3)
TO AH-D-14A→D

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	AK2.07 (068)	
	Importance Rating	3.3	3.4

(K&A Statement) Knowledge of the interrelations between the Control Room Evacuation and the following: ED/G

Proposed Question: Common 60

Plant conditions:

- The Control Room has just been evacuated due to a fire.
- OP-TM-EOP-020, "Cooldown From Outside the Control Room", Immediate Manual Actions were completed prior to leaving the Control Room.
- All Remote Transfer switches have been operated.

Event:

- A Loss of offsite power (LOOP) occurs.

With the above conditions....

A. "A" and "B" Emergency Diesels will start automatically and re-energize their respective bus.
B. "A" and "B" Emergency Diesels will start automatically but their respective breakers must be closed locally.
C. "A" Emergency Diesel must be started manually from the 1D 4160 Volt Bus and its breaker closed locally.
D. "B" Emergency Diesel will have to be started manually from the 1E 4160 Volt Bus and its breaker closed locally.

Proposed Answer: D. "B" Emergency Diesel will have to be started manually from the 1E 4160 Volt Bus and its breaker closed locally.

Explanation (Optional):

A. Plausible since it is correct for "A" EDG.
B. Plausible since the B Diesel breaker must be closed locally.
C. Plausible since it is correct for "B" EDG.
D. Correct – EG-Y-1B Emergency Bypass Selector is placed in BYPASS in EOP-20, Attachment 10 as part of establishing control at the RSD.

Technical Reference(s): OP-TM-EOP-020, Cooldown From Outside the Control Room (Page 52) (Attach if not previously provided)

EOP-020 Basis Step 3.1.2, Page 15&16

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.7

55.43 ____

Comments:

ATTACHMENT 7
Starting EG-Y-1B and Loading 1E 4160V Bus
(Performed at 1E 4160V bus)
Page 1 of 1

WARNING

Relay Room CO2 actuation may have occurred. Be cautious upon entering ESAS room, 1E 4160V or 1D 4160V rooms. A wintergreen odor indicates that the environment may be IDLH.

1. **TRIP** the following breakers on 1E 4160V bus
   - 1SA-E2 ______
   - RR-P-1B ______
   - BS-P-1B ______
   - DH-P-1B ______
   - EF-P-2B ______
   - T1-E2 ______
   - 1SB-E2 ______
2. Start and load EG-Y-1B (at Unit 1E3)
   - 2.1 **PUSH** "START" for EG-Y-1B ______
   - 2.2 **VERIFY** "Ready" light is lit ______
   - 2.3 **UNLOCK and PLACE** "Feeder Transfer" 69 switch in "Emergency" ______
   - 2.4 **NOTIFY** operator at RSD panel ______
   - 2.5 **PUSH** "CLOSE" for G11-02 ______
3. Start EF-P-2B
   - 3.1 **NOTIFY** operator at RSD panel "EF-P-2B will be started" ______
   - 3.2 **PUSH** "CLOSE" for EF-P-2B (Unit 1E5) ______
4. Re-energize 'B' ES power train
   - 4.1 **NOTIFY** operator at RSD panel ______
   - 4.2 **PUSH** "CLOSE" for S1-02 (Unit 1E6) ______
   - 4.3 **PUSH** "CLOSE" for T1-02 (Unit 1E12) ______

Step 3.1.1 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 Safe Shutdown AO goes to the EFW area, and Primary Safe Shutdown 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 Personnel hatch and Primary AO Central.

PURPOSE:

Provide information on personnel response locations, available communications systems, and location of portable lighting.

BASIS:

Information on personnel response locations is provided to ensure that personnel report to appropriate areas for initial required actions. Information on M&I phones is provided to ensure personnel are aware of communications systems which will remain available during the fire. Information on portable lighting locations is provided to ensure personnel are aware of storage location if this lighting is needed.

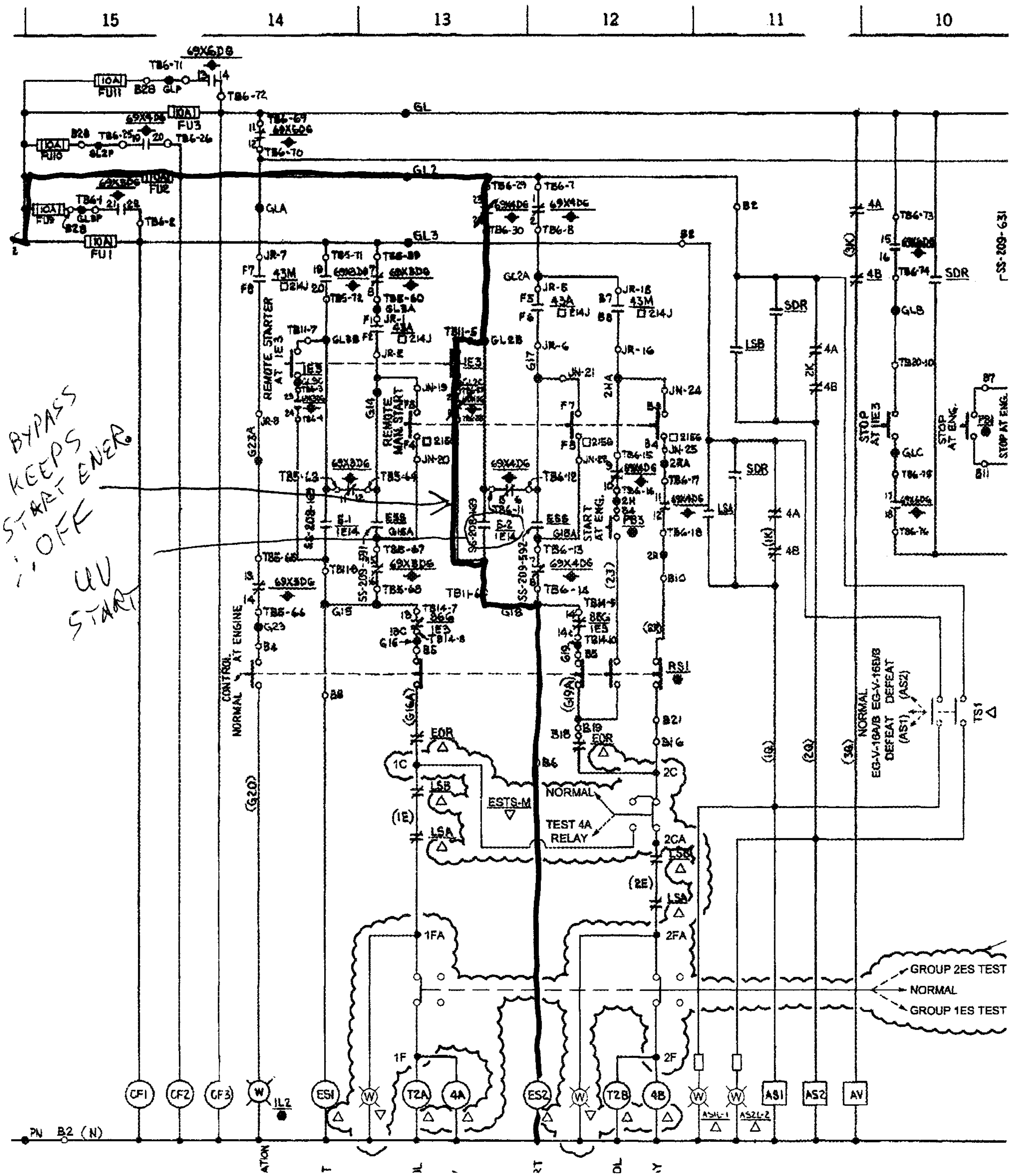
Step 3.1.2 **PERFORM** Attachment 10 **and** Attachment 9 to transfer control to the RSD panels.

PURPOSE:

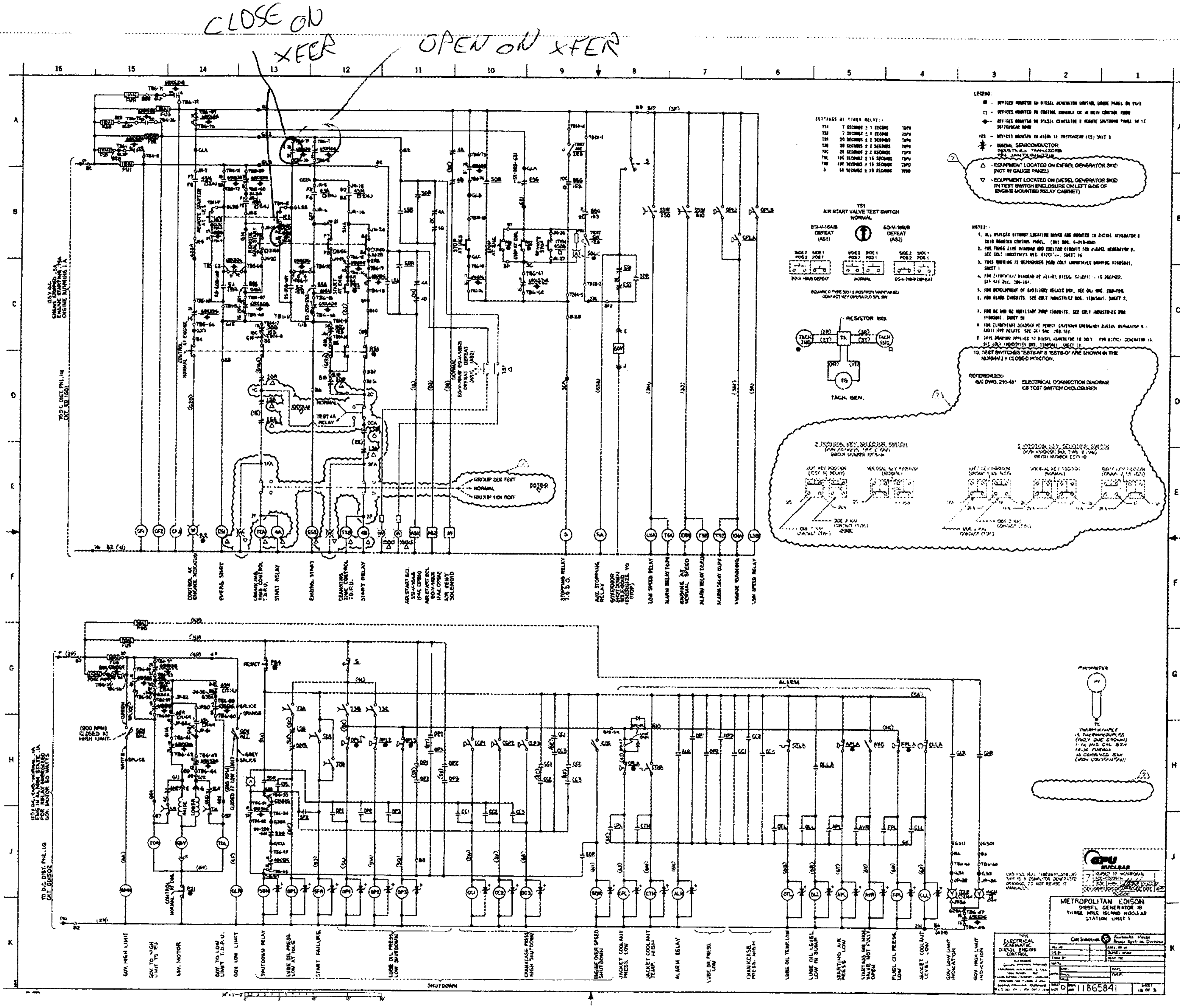
Transfer control of plant equipment to the RSD Panels.

BASIS:

A Control Room evacuation will require equipment to be controlled at an alternate location. A fire in the Control Room or Relay Room could affect control circuits for required equipment, resulting in inoperability or spurious operation. Transferring control of Safe Shutdown equipment isolates control circuits subject to fire damage, and provides the ability to control the equipment from the RSD Panels or locally.



15
14
13
12
11
10
REMOTE STARTER AT 1E3
REMOTE MAN. START
START AT ENG.
STOP AT 1E3
STOP AT ENG.
CONTROL AT ENGINE NORMAL
NORMAL EG-V-16A/B EG-V-16B/B DEFEAT DEFEAT (AS1) (AS2)
TS1
ESTS-M
NORMAL
TEST 4A RELAY
GROUP 2ES TEST
NORMAL
GROUP 1ES TEST
PN B2 (N)
BYPASS KEEPS START ENER. ; OFF
UV START


CLOSE ON XFER
OPEN ON XFER
TS1
AIR START VALVE TEST SWITCH
NORMAL
RESISTOR BOX
TACH. GEN.
START RELAY
ENGINE START
EMERG. START
STOPPING RELAY
LOW SPEED RELAY
ALARM RELAY
ALARM
SHUTDOWN
SHUTDOWN RELAY
START FAILURE
GOV. HIGH LIMIT
GOV. LOW LIMIT
ENGINE OVER SPEED SHUTDOWN
JACKET COOLANT PRESS. LOW
JACKET COOLANT TEMP. HIGH
LUBE OIL PRESS. LOW
CRANKCASE PRESS. HIGH
LUBE OIL TEMP. LOW
LUBE OIL LEVEL LOW IN SUMP
STARTING AIR PRESS LOW
FUEL OIL PRESS. LOW
JACKET COOLANT LEVEL LOW
GOV. LOW LIMIT INDICATION
GOV. HIGH LIMIT INDICATION
METROPOLITAN EDISON
DIESEL GENERATOR 1B
THREE MILE ISLAND NUCLEAR STATION UNIT 1
ELECTRICAL SCHEMATIC DIESEL ENGINE CONTROL
GPU NUCLEAR
11865841

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	AA1.2 (A01)	
Importance Rating	3.2	3.5

(K&A Statement) Ability to operate and / or monitor the following as they apply to the (Plant Runback): Operating behavior characteristics of the facility.

Proposed Question: Common 61

Plant conditions:

- Reactor power was 100% when a dropped rod in Group 3 occurred a few minutes ago.
- Reactor power is currently 62% and stable.
- Turbine load 532 MWe and stable.
- MAP Alarm H-1-1, "ICS RUNBACK", is illuminated.
- MAP Alarm H-2-1, "ICS IN TRACK", is illuminated.
- MAP Alarm H-3-4, "ULD HIGH LOAD LIMIT", is illuminated.

Given the above conditions the operator will have to....

A. reduce the ULD output until NI power is <55%.
B. reduce the ULD output until NI power is <60%.
C. take Hand control of the SG/RX Master and reduce NI power to <55%.
D. take Hand control of the Diamond and Feedwater stations to reduce power to <60%.

Proposed Answer: C. take Hand control of the SG/RX Master and reduce NI power to <55%.

Explanation (Optional):

A. Plausible since <55% is the target power level; however the ULD is not operable since the ICS is still in track.
B. Plausible since <60% is the Tech Spec limit for an inoperable rod after a time limit; however the ULD is not operable since the ICS is still in track.
C. Correct answer. The ULD Substation is inoperable so the SG/Rx Master will have to be used to lower power below the Runback limit.
D. Plausible since the Diamond and Feedwater Substations could be used to reduce power; however power has to be reduced to <55%.

Technical Reference(s): OP-TM-MAP-H0304, ULD High Load Limit (Attach if not previously provided)
OP-TM-H0201, ICS in Track
OP-TM-MAP-H0101, ICS Runback
OP-TM-621-471 ICS Manual Control

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 B.7
55.43 ______

Comments:

1.0 SETPOINTS

- ULD > Hi Load Limit (MWe limit)

2.0 CAUSES

- Loss of one RCP (665 MWe, ≈ 75% power).
- Loss of two RCPs (405 MWe, ≈ 50% power).
- Loss of one MFP (560 MWe, ≈ 68% power).
- Asymmetric rod condition (455 MWe, ≈ 55% power).
- Degraded RCS flow* (Variable)
- Manual Hi Load Limit set below ULD demand (Variable)

3.0 AUTOMATIC ACTIONS

NOTE: ULD demand becomes the most limiting of the above listed load limits

- Auto runback to associated limit at the rate of:
  - 50%/minute for a loss of RCP or MFP.
  - 30%/minute for an Asymmetric/dropped rod.
  - 20%/minute for degraded RCS flow.
  - % selected on Load Rate of Change dial for Manual Hi Load Limit.
- ULD station goes to track (both white HAND and red AUTO lamps Lit)

4.0 MANUAL ACTIONS REQUIRED

- **ENSURE** NI power and feedwater are lowering at the rate specified above.
- **If** cause of alarm is the manual High Load Limit setpoint, **then VERIFY** proper setpoint **and REDUCE** demand on Unit Load Demand station to below setpoint.

1.0 SETPOINTS

- ULD tieback (mini-track) in effect

2.0 CAUSES

- High load limit
- Low load limit
- ICS in track
  - Reactor tripped
  - Turbine generator breakers open
  - Cross limits
  - SG/Reactor Demand station in HAND
  - Turbine in manual
    (DTCS in "LOCAL-OWS" control or
    ICS Main Turbine station in HAND)
  - Both SG A(B) FW Demand stations in HAND.
  - Reactor Demand station in HAND
  - Diamond Panel in Manual

3.0 AUTOMATIC ACTIONS

- ULD uses high load limit, low load limit or
  generated megawatts as the integrated demand signal.

4.0 MANUAL ACTIONS REQUIRED

- **VERIFY** plant control in balance by observing the following:
  - RCS T_{avg}
  - Turbine Header Pressure
  - RCS Pressure
  - FW Flows
  - Reactor (NI) Power
  - MWe

- **IAAT** MAP H-2-1 returns to normal, **then**
  - **VERIFY** ULD transfers to HAND
  - **ENSURE** ULD output is stable and below setpoint (A5001)

1.0 SETPOINTS

- Loss of one RCP and > 665 MWe
- Loss of two RCP and > 405 MWe
- Loss of one main feed pump and > 560 MWe
- Asymmetric rod fault and > 455 MWe.
- RC flow limit: variable
  MWe equivalent to percent core power $\geq \frac{105 \times \text{Actual RC flow}}{140 \times 10^6 \text{ lbs/hr}}$

(at ~135×10^6 lbs/hr, RC flow starts to limit power at 100%)

2.0 CAUSES

- RCP Trip
- MFP Trip
- Dropped Rod
- RCS Flow Degradation

3.0 AUTOMATIC ACTIONS

NOTE: Actual reactor power may vary due to plant efficiencies.

NOTE: ULD STAR module will trip to Manual following runback.

- Unit runs back to:
  - 665 MWe (≈ 75% NI power) at a rate of 50%/minute on a loss of 1 RCP.
  - 405 MWe (≈ 50% NI power) at a rate of 50%/minute on a loss of 2 RCPs.
  - 560 MWe (≈ 68% NI power) at a rate of 50%t/minute on a loss of 1 MFP.
  - 455 MWe (≈ 55% NI power) at a rate of 30%/minute on an Asymmetric rod fault.
  - RC flow limited MWe power level at a rate of 20%/minute on RC Flow degradation.

4.0 MANUAL ACTIONS REQUIRED

- **ENSURE** NI power is reduced to below the limit for the runback condition.
  - **If** ICS manual control is required, **then INITIATE** OP-TM-621-471, ICS Manual Control.
  - **IF** there is a dropped rod, **then GO TO** OP-TM-AOP-062, Inoperable Rod.
- **INITIATE** 1102-4 for power reduction
- **DOCUMENT** required entries in transient cycle logbook.

Exelon Nuclear

ICS MANUAL CONTROL

1.0 PURPOSE

1.1 Provide direction for control of reactor power and feedwater flow with (1) SG/Rx Demand in HAND, (2) SG FW A/B DEMAND stations in HAND or (3) Diamond control in Manual.

2.0 MATERIAL AND SPECIAL EQUIPMENT - None

3.0 PRECAUTIONS, LIMITATIONS, AND PREREQUISITES

3.1 Precautions:

3.1.1 To avoid CRDM stator damage from overheating, do **not** move any CRDM > 420" (or 300%) in one hour.

3.1.2 To avoid CRDM stator damage from overheating, do **not** move any CRDM > 30 minutes in one hour.

3.2 Limitations

3.2.1 Maintain NI power within 1% of desired reactor power.

3.2.2 Maintain control rod index above error adjusted rod index limit for number of RCPs operating IAW COLR Figure 1, 2 or 3.

3.2.3 Maintain OTSG level 23 to 27 inches when reactor power < 20%.

1. Except as required by GOPs or EOPs when reactor is shutdown.

3.2.4 When reactor power > 20%, then control feedwater flow to OTSGs as follows:

- Maintain T_{avg} between 578°F and 580°F.
- If T_{avg} reduced at EOC IAW 1102-4, then maintain T_{avg} within ±1°F of desired setpoint.
- Maintain ΔT_C between +2°F to –2°F ($\Delta T_C = T_C A - T_C B$).
- If only 3 RC Pumps are operating, then when either OTSG level reaches 25", ΔTc limit does not apply

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	AK3.2 (A03)	
Importance Rating	3.0	3.5

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the (Loss of NNI-Y): Normal, abnormal and emergency operating procedures associated with (Loss of NNI-Y).

Proposed Question: Common 62

Which ONE of the choices completes the following statement regarding operator action on Loss of ICS Hand Power?

The reactor must be tripped if reactor power is ______________.

A. < 75% and FW-V-17A is in HAND because a feedwater transient will occur.
B. < 75% and both FW-16A and FW-V-16B are in HAND because an RCS overcooling will occur.
C. > 75% and "A" Feedwater Pump is in HAND because an RCS heatup will occur.
D. > 75% and Turbine Bypass Valves are in HAND because reactor power will exceed 100% power.

Proposed Answer: A. < 75% and FW-V-17A is in HAND because a feedwater transient will occur.

Explanation (Optional):

A. Correct answer. With power less than 75% and the Feedwater Valve in HAND, a feedwater upset will occur.
B. Plausible since these valves will fail to mid-position; however this would be a reduction in feedwater and the FW-V-17s will compensate
C. Plausible if student believes pump would go to zero speed on loss of ICS auto, since an RCS heat-up could occur if Feedwater flow was reduced; however the Feedwater valves and remaining pump will compensate for FW Pump speed changes.
D. Plausible since the Turbine Bypass Valves would fail open and tend to raise power; however with the ICS in AUTO the Reactor subsystem will compensate.

Technical Reference(s): OP-TM-AOP-0261, Loss of ATB or ICS Hand Power Basis Document (Page 4) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.10
55.43 ______

Comments:

NRC Q'n Common G2

In the event of a loss of a non ICS HAND power sub-feed without loss of ATB or ICS/NNI HAND power, the CRS may use this procedure to identify appropriate actions. In the event of an emergency (e.g. fire) that requires an unplanned loss of ATB or ICS HAND power, the CRS may direct actions in this procedure before ATB or HAND power is de-energized.

4.0 IMMEDIATE ACTIONS

4.1 **"IAAT** reactor power **< 75% and** either FW-V-17A **or** FW-V-17B is in Hand"
If reactor power is less than 75%, and a Main FW Reg valve is in hand, there will be a significant FW upset, without the ability to independently adjust FW flow to the OTSGs.

4.2 This IAAT condition is applicable once the procedure is entered until Hand power is restored. If a FW-V-17A or B is in Hand and a reactor TRIP occurs after entering AOP-26, then this condition applies.
If the reactor was at hot shutdown when hand power is lost, and FW valves are in Hand, then this condition applies. (i.e. TRIP both main FW pumps and INITIATE EFW.)

4.3 When FW control is not adequate (i.e. IAAT condition applies),
PERFORM EOP-001 IMA (Trip reactor and turbine, and perform any contingencies),
TRIP both Main FW pumps
INITIATE EFW
INITIATE EOP-001 Symptom Check & VSSV

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	AK2.2 (A04)	
Importance Rating	3.3	3.5

(K&A Statement) Knowledge of the interrelations between the (Turbine Trip) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: Common 63

Plant conditions:

- The plant is starting up.
- Reactor power is 42%.
- Tave is 579 °F.
- Turbine load is 360 MWe.

Event:

- Turbine trip, on stator coolant loss.

Which ONE of the following is the required operator action?

A. Verify an automatic reactor trip on interlock with the turbine trip.
B. Ensure reactor power is reduced to within Turbine Bypass Valve capability.
C. Verify an ICS stator coolant runback is in progress to reduce power.
D. Initiate a MANUAL reactor trip if any Main Steam Safety Valves opens.

Proposed Answer: B. Ensure reactor power is reduced to within Turbine Bypass Valve capability.

Explanation (Optional):

A. Plausible since an automatic reactor trip will occur at > 45% power.
B. Correct – the turbine trip will not cause a reactor trip at < 45% power and no auto runback will actuate.
C. Plausible since an ICS runback does reduce power however it is from track not stator coolant runback which only affects turbine.
D. Plausible since this is an action on a steam leak and MSSV's are likely to open during this event.

Technical Reference(s): MAP Alarm K-1-1 Turbine Trip (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 b.5
55.43 ______

Comments:

- Moisture sep. level hi-hi (after 12 second delay) (<u>any</u> 2 of 6 Hi-Hi level switches or one level switch and one other Intercept Valve already closed)
- Reactor trip
- Generator fault
- Manual trip - front standard trip handle or control room pushbutton

3.0 <u>AUTOMATIC ACTIONS</u>

- Turbine trip
- Turbine Intercept, Intermediate Stop, Main Stop, and Control Valves Close
- Reactor trip (if Reactor power level >45 percent)

4.0 <u>MANUAL ACTIONS REQUIRED</u>

4.1 **If** Reactor has tripped, **then GO TO** OP-TM-EOP-001, Reactor Trip.

4.2 **ENSURE** Reactor power is reduced IAW 1102-4 Power Operations to allow the Turbine bypass valves to control OTSG pressure.

4.3 **PERFORM** OP-TM-301-151 "MAIN TURBINE GENERATOR OPERATING MODE TO STANDBY MODE".

4.4 **REVIEW** transient cycle logbook and make <u>all</u> required entries.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EK2.2 (E03)	
	Importance Rating	4.3	4.3

(K&A Statement) Knowledge of the interrelations between the (Inadequate Subcooling Margin) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: Common 64

Plant conditions:

- Reactor was tripped due to a large RCS leak (LOCA).
- EOP-001, "Reactor Trip", was initiated.
- Subcooling margin is 12 °F.
- OP-TM-EOP-010 Rule 1, "Loss of Subcooling Margin (SCM)", has been initiated.
- RC-P-1D was not tripped within one minute and is currently running.

Which ONE of the following verified conditions allows RC-P-1D to be stopped?

A. Tclad > 1800 °F.
B. Adequate HPI flow.
C. "A" LPI Flow 1400 gpm, "B" LPI Flow 900 gpm.
D. Both OTSG levels are between 75 to 85% Operating Range Level.

Proposed Answer: A. Tclad is >1800 °F.

Explanation (Optional):

A. Correct answer. Rule 1, Step 2 RNO column.
B. Plausible since HPI is verified; however it is not one of the criteria for securing the RCP.
C. Plausible since >1250 gpm is the correct value; however it must exist in both loops.
D. Plausible because it is a step in RULE 1 but not RC-P stop criteria.

Technical Reference(s): OP-TM-EOP-010, Rule 1 Loss of Subcooling Margin (SCM) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	B.10	
	55.43		

Comments:

NRC Q4 Comment # 9

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 in each line - Tclad > 1800°F
3. **INITIATE** OP-TM-642-901 "1600 # ESAS ACTUATION"	
4. **INITIATE** EFW IAW Guide 15 **and FEED** available OTSGs to 75 to 85% Operating Range Level.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EK3.3 (E09)	
	Importance Rating	3.8	3.4

(K&A Statement) Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Cooldown): Manipulation of controls required to obtain desired operating results during abnormal and emergency situations.

Proposed Question: Common 65

Plant conditions:

- Reactor Tripped due to loss of offsite power (LOOP).
- Both Emergency diesels started and are supplying their respective busses.
- All three Emergency Feedwater Pumps have started.
- Natural Circulation flow has been verified.
- T cold is 520°F

Which ONE of the following describes the method of cooldown, the cooldown rate limit, and the reason for that limit?

A. Operate the Atmospheric Dump Valves to control cooldown rate ≤ 30 °F/hr to prevent voiding in the hot legs.
B. Operate the Atmospheric Dump Valves to control cooldown rate ≤ 50 °F/hr to prevent formation of a head bubble.
C. Operate the Turbine Bypass Valves to control cooldown rate ≤ 30 °F/hr to prevent voiding in the hot legs.
D. Operate the Turbine Bypass Valves to control cooldown rate ≤ 50 °F/hr to prevent formation of a head bubble.

Proposed Answer: B. Operate the Atmospheric Dump Valves to control cooldown rate ≤ 50 °F/hr to prevent formation of a head bubble.

Explanation (Optional):

A. Plausible because the first part is correct.
B. Correct – Turbine Bypass Valves are not available during a LOOP and the NC cooldown rate limit is ≤ 50 °F/hr to prevent formation of a head bubble.
C. Plausible because it is the limit with Tave < 255 °F and is the normal means of cooldown with off-site power available.
D. Plausible because the limit is correct and is the normal means of cooldown with off-site power available.

Technical Reference(s): OP-TM-EOP-010, Guide 11, Cooldown Rate Limits (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 B.10

55.43 ______

Comments:

Guide 11
Cooldown Rate (CDR) Limits

IAAT reactor is shutdown **and** SCM > 25°F, **then** the RCS cooldown rate limit is:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** RCS temperature > 255°F.	RCS Cooldown rate limit is 30°F/HR. **EXIT**
2. **VERIFY** at least one RCP is operating.	RCS Cooldown rate limit is 50°F/HR **EXIT**
3. **VERIFY** OTSG tube leak rate < 1 GPM	**If** OTSG isolation is required **and** RCS pressure > 1000 psig **and** RCS temperature > 500F **then** RCS Cooldown rate limit is 240°F/HR
4. RCS Cooldown rate limit is 100°F/HR.	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	G2.1.17	
Importance Rating	3.5	3.6

(K&A Statement) Ability to make accurate, clear and concise verbal reports.

Proposed Question: Common 66

Which ONE of the following is an acceptable communication during an emergency condition?

A. Announcing to control room personnel "RM-A-5 and RM-A-15 are in alarm" during the performance of reactor trip immediate manual actions.
B. Announcing on the plant page "Fire in the Auxiliary Building, Elevation 305".
C. During an update the CRS announces "Attention for a brief".
D. Giving an update to the crew "control rods are at 290% and inserting and reactor power is at 101% and rising".

Proposed Answer: D. Giving an update to the crew "control rods are at 290% and inserting and reactor power is at 101% and rising".

Explanation (Optional):

A. Plausible since this is a condition to be announced; however it should not be done during the performance of the reactor trip IMAs.
B. Plausible since an announcement is required but this does not meet AOP-001, FIRE criteria.
C. Plausible since updates and briefs are common communications activities but an update is a higher priority than a brief.
D. Correct – management expectation for identifying specific indications, responses, and trends.

Technical Reference(s): OS 24, Conduct of Operations During Abnormal and Emergency Events (Pages 18-21) (Attach if not previously provided)

HU-AA-101 Pg 10

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank #

Modified Bank # 23519 (Note changes or attach parent)

		(INPO)
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

NRC QU Common 66

4.4.2. For non-face-to-face verbal communication, the sender and receiver shall **IDENTIFY** themselves by stating their name or title.

4.4.3. **USE** Phonetic alphabet (shown in Attachment 3), as required, to ensure proper component identification.

1. **USE** 3-way communications for all information exchanges that will result in decision making, direction being given or actions being taken..

4.4.4. **AVOID** words during verbal communication that could be mistaken for each other, such as "increase" and "decrease".

4.4.5. Communication of indicator readings should be provided in the format of PARAMETER – VALUE – TREND (with rate when appropriate).

FOR EXAMPLE

Reactor pressure is 1000 psig and going down.

4.4.6. **AVOID** the use of sign language.

4.4.7. **USE** the appropriate unit designator, system designator, or noun name and appropriate phonetic alphabet component or train designator when communicating equipment nomenclature.

FOR EXAMPLE

1MS009A should be verbalized as "one MS zero zero nine alpha"

FOR EXAMPLE

HV-2-10-71A should be verbalized as "H V two ten seventy-one alpha"

4.4.8. Sender Responsibilities (Direction)

1. **ADDRESS** intended receiver by name or title.
2. **SPEAK** clearly **and OBTAIN** the attention of the intended receiver.
3. **SEND** the intended message in just enough words to minimize the chance of receiver misunderstanding.
4. **REQUIRE** confirmation from the intended receiver of the information.

NRC QM Comment 66

4.3.5 Reactor Trip OP-TM-EOP-1 Actions

1. When OP-TM-EOP-1, Reactor Trip is entered, Reactor Trip Immediate Manual Actions take priority over Immediate Manual Actions in any other EVENT PROCEDURE.

2. When a reactor trip is required or occurs, the first team member performs an UPDATE to announce "Reactor Trip" to the Control Room team.

3. The control room supervisor acknowledges entry into the Reactor Trip procedure by directing performance of immediate manual actions.

4. The Unit Reactor Operator verbalizes and performs the Immediate Manual Actions from memory.

   Any procedural contingency action for Immediate Manual Actions, if applicable, is verbalized and performed by the Unit Reactor Operator without delay, and does not require concurrence from the Control Room Supervisor.

   Examples:

   - If the turbine fails to trip, both EHC pumps are placed in the "Pull To Lock" position and an operator is sent to open EHC-V-FV1.

   - If the reactor is not shutdown, 1L-02 & 1G-02 are OPENED, HPI is initiated, and primary to secondary heat transfer is maintained.

5. While the URO performs the IMAs, the Assistant Reactor Operator (ARO) performs a SYMPTOM CHECK.

6. The Control Room Supervisor verifies completion of Reactor Trip immediate manual actions and any contingency action performed.

7. If a loss of Subcooling Margin (SCM Rule 1) or Excessive Heat Transfer (XHT Rule 3) is identified during the performance of Reactor Trip Immediate Manual Actions, the crew member identifying the symptom announces the symptom to the team, the applicable Rule is pulled, concurrence is obtained from the Control Room Supervisor, and the Rule is immediately performed by the Assistant Reactor Operator with the following priority of Rule implementation:

   - Loss of Subcooling Margin (SCM Rule 1)

   - Excessive Heat Transfer (XHT Rule 3)

8. Upon completion of Reactor Trip Immediate Manual Actions, the Unit Reactor Operator performs the following:

    1. Initiates "Global Silence" of Annunciator alarms.

    2. Announces "Global Silence" activated and "Reactor Trip Immediate Manual Actions complete".

9. The Control Room Supervisor acknowledges initiation of "Global Silence", and requests the results of the SYMPTOM CHECK from the Assistant Reactor Operator.

10. The ARO notifies the team of <u>any</u> symptoms identified, and <u>any</u> Rules performed or in progress.

    Communication example:

    "Adequate subcooling margin exists, subcooling margin is 58 °F and rising slowly. There is no evidence of excessive or lack of primary to secondary heat transfer based on indicated RCS temperature, OTSG levels and OTSG pressures. Radiation monitoring does not indicate <u>any</u> OTSG tube leakage."

11. The Control Room Supervisor UPDATES the team of transfer to the Vital System Status Verification (VSSV) section of OP-TM-EOP-1 or to another EOP if required based on results of the SYMPTOM CHECK.

12. Reactor Operators continue to periodically perform SYMPTOM CHECK.

13. When a reactor trip or turbine trip is announced, then the auxiliary operators go to the stations identified on Attachment E. The AO at the assigned stations will take the following actions per Attachment E immediately (i.e. w/o CR concurrence) and NOTIFY the control room when the action is complete.

    - "Fire the Aux Boiler" per OP-TM-414-401(402)

    - OPEN AS-V-8 to make aux steam available to Gland Steam when aux steam pressure is available

    - Any other actions will be performed at CRS direction or with CRS concurrence.

4.3.6 Reactor "At Power" Protocol

A. Any time an automatic control or interlock functions fails to perform as designed and there is no specific procedure direction, the reactor operator should take action to compensate directly for the failure. The reactor operator should verbalize the actions taken to inform the control room team of the condition and actions taken. (e.g. Pressurizer level < setpoint and MU-V-17 is closed in automatic. The RO should place MU-V-17 in hand, adjust makeup flow and announce the condition to the control room.)

B. Following any plant upset, reactivity and primary to secondary heat transfer parameters are reviewed and the reactor operator provides feedback on plant conditions. That communication should be modeled on the following example:

"The runback is complete. The plant is stable at 55% reactor power. Feedwater flow rate and generator output correspond to reactor power."

B. During a planned or forced plant power maneuver, a PLANT ANNOUNCEMENT should be made approximately every 10% change in reactor power.

C. If plant control becomes unstable, then Reactor power should not be raised to stabilize plant conditions. Adjust feedwater and header pressure or lower reactor power.

4.3.7 Manual Control of Systems

A. Manual control of a system may be required or directed for any of the following reasons:

- Failure of automatic systems to perform or respond correctly
- Procedural action to control a parameter (i.e. SCM)

B. When manual control is initiated, the performer notifies the Unit team of the action.

C. Control bands are requested by the performer, and Shift management provides appropriate control parameters.

4.4 Communication

4.4.1 The communication standards of OP-AA-104-101 apply. Three way communication is used for direction, commands, and transfer of technical information used as the basis for plant control.

4.4.2 The following communication example applies to any EVENT PROCEDURE that contains Immediate Manual Action(s):

- When EVENT PROCEDURE entry is recognized, an UPDATE is used

- The CRS acknowledges entry (if CRS does not make UPDATE) and directs immediate manual actions.
- The reactor operator VERBALIZES immediate manual actions performed
- The CRS acknowledges completion of the immediate manual actions

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 any team members have questions.

E. A brief ends with the statement "End of BRIEF".

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G2.1.27	
	Importance Rating	2.8	2.9

(K&A Statement) Knowledge of system purpose and or function.

Proposed Question: Common 67

Which ONE of the choices completes the following statement?

The Containment Hydrogen Monitor is placed in service __________ and the alarm setpoint is __________.

A. prior to entering HOT STANDBY; 2%
B. prior to entering HOT STANDBY; 4%
C. when directed by emergency operating procedures; 2%
D. when directed by emergency operating procedures; 4%

Proposed Answer: C. when directed by emergency operating procedures; 2%

Explanation (Optional):

A. Plausible since post-accident monitoring instrumentation is required to enter HSB. Alarm setpoint is correct.
B. Plausible since post-accident monitoring instrumentation is required to enter HSB and 4% is a well-known number just below the flammability limit. However, 4% is beyond the point of hydrogen recombiner and H2 purge operation.
C. Correct. Containment Hydrogen Monitor is maintained in standby until alignment for operation is specified by procedure or directed by management.
D. Plausible because the first part is correct.

Technical Reference(s): 1105-18, Page 3 (2.3) (Attach if not previously provided)
MAP NN-1-8 "RB H2 conc. Hi"

Proposed references to be provided to applicants during examination: None

Learning Objective: __________ (As available)

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

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 b.9
55.43 ________

Comments:

1.0 **REFERENCES**

1.1 Operation Manual for k III containment Hydrogen Monitor by Comsip, Inc.

1.2 Post Accident Hydrogen Monitoring System C-302-674

2.0 **LIMITS AND PRECAUTIONS**

2.1 Equipment

1. Limits

		Minimum	Maximum
a.	Temperature	300°F	260°F
b.	Pressure	60 psig	-5 psig
c.	Humidity	100 percent	0
d.	Radiation	10^6 Rads. (total)	0

2. Keys #4 and #140 from controlled key locker in Control Room.

2.2 Personnel

1. In the event of a tubing system leak the inside of the panel may become filled with radioactive, explosive or very hot, (greater than 270°F), gases. Exercise extreme caution when entering the panel.

2. The heated sample compartment is maintained at approximately 275°F when the analyzer system is either in the "Stand-by" or "Analyzer" mode. In order to prevent injury this compartment should be allowed to cool for a minimum of four hours before performing maintenance inside or closely surrounding this area.

3. There are many exposed terminals in both the analyzer and the remote panels where personnel could come in contact with high voltage, (120VAC). The circuit breaker in the analyzer panel should always be in the off position before any maintenance is attempted.

2.3 Administrative

1. Because the H_2 Monitor is normally maintained in the standby condition (isolated from containment but with the electronics kept warm), the control room chart recorder drive will remain off and will be turned on when the monitor is placed in service.

2. Computer points A0425 & A0904 (High Containment H_2) may be placed in "Deleted from Scan" to avoid nuisance alarms. These alarms will be verified to be in "Returned to Scan" whenever the monitor is placed in service.

3. If taking a hydrogen monitor out of service, refer to Tech. Spec. 3.5.5.2.

NN-1-8
Revision 4

ALARM:

RB H_2 Conc Hi

SET POINTS:

Channel "A" RB H_2 Analyzer 2 percent
Channel "B" RB H_2 Analyzer 2 percent

CAUSES:

Loss of Coolant Accident
Calibration of H_2 Mon. in progress
Venting pressurizer to containment

AUTOMATIC ACTION:

None

OBSERVATION (CONTROL ROOM):

Check H_2 Analyzer Recorders on PL

MANUAL ACTION REQUIRED:

Startup Hydrogen Recombiner per 1104-62. Draw sample to confirm hydrogen value.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.34	
	Importance Rating	2.8	3.2

(K&A Statement) Knowledge of the process for determining the internal and external effects on core reactivity.

Proposed Question: Common 68

Plant conditions:

- The reactor tripped 24 hours ago.
- Core Burnup is 300 EFPD
- An Estimated Critical Position calculation was completed with:
  - RC Tave = 532 °F
  - RC Pressure = 2155 psig

Current conditions:

- RC Tave = 531 °F
- RC Pressure = 2120 psig

Which ONE of the choices completes the following statement?

During the startup, criticality will occur______ in the rod band than the ECP because there is a net ______ reactivity change.

A. lower; negative
B. lower; positive
C. higher; negative
D. higher; positive

Proposed Answer: B. lower; positive

Explanation (Optional):

A. Plausible because the first part is correct.
B. Correct: facility thumb-rule is 100 psi = 1°F.
C. Plausible because it balances the choices.
D. Plausible because the second part is correct.

Technical Reference(s): Lesson Plan 11.2.01.124, Reactivity Coefficients, Page 6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.1

55.43 ____

Comments:

INSTRUCTOR NOTES

TP-2

1. Moderator Temperature Coefficient

   a. Reactivity change per unit moderator temperature change

   b. $\alpha_{MT} = \Delta k/k/°F$ (mod)

Obj. 8.01

   c. Cycle 12 values: 10^{-4} $\Delta k/k/°F$ (Mod) at Hot Full Power.

2. Moderator Pressure Coefficient

TP-3

   a. Reactivity change per unit moderator pressure change.

   b. $\alpha_{MP} = \Delta k/k/100$ psig

Obj. 8.01

   c. Cycle 12 values: 10^{-4} $\Delta k/k/100$ psig at Hot Full Power.

   d. **Note that at Hot Full Power conditions, 1°F or 100 psig results in the same moderator density change.**

3. Moderator Void Coefficient

TP-4

   a. Reactivity change per present moderator void.

   b. $\alpha_v = \Delta k/k/\%$ void

Obj. 8.01

   c. Value (approx): -3.0 to 19.0 x 10^{-4} $\Delta k/k/\%$ VOID.

TP-4B - FSAR 3.2.5

   d. Coefficient quoted above is for a core just beginning to void. Void coefficient is strongly influenced by boron concentration and existing voids.

   e. At power, void coefficient ALWAYS negative in U.S. power reactors, and U.S. weapons production reactors (unlike Chernobyl).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2.2.26	
	Importance Rating	2.5	3.7

(K&A Statement) Knowledge of refueling administrative requirements.

Proposed Question: Common 69

Which ONE of the following describes a VIOLATION of refueling administrative requirements?

A. Fuel moves in Spent Fuel Pool are being supervised by an individual with an inactive SRO license.
B. An irradiated fuel assembly has been transferred to the Spent Fuel Pool with both trains of ESF Ventilation secured.
C. The fuel grapple FULL DOWN position has been determined by ZZ tape reading rather than by the Digital Fuel Elevation reading.
D. The Main Fuel Bridge has been left unattended with the fuel grapple at GRAPPLE UP DISENGAGED and the bridge de-energized.

Proposed Answer: B. An irradiated fuel assembly has been transferred to the Spent Fuel Pool with both trains of ESF Ventilation secured.

Explanation (Optional):

A. Plausible because an active license applies for core geometry changes but not spent fuel movement in the FHB.
B. Correct – ESF Ventilation is required to be in operation per Administrative L&P 1507-7, 5.2.2.
C. Plausible because both are methods for determining grapple position but ZZ tape meets the administrative requirement.
D. Plausible because properly securing the bridge is an administrative requirement. However, this is an acceptable method per 1507-3.

Technical Reference(s): 1507-7, 5.2.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

NOTE

Overload and Underload warning lights may momentarily flicker as the winches start, stop, or change speed or when the transfer carriage spans a gap in the track.

5.1.2 If any warning or limit condition light comes on, operations shall be stopped immediately and the Fuel Handling Supervisor or FHBS shall be notified immediately.

5.1.3 No interlocks shall be bypassed except as specified by approved procedure. All such bypasses of interlocks shall be entered in the Control Room Log.

5.1.4 The fuel transfer tube isolation valve (FH-V-1A and B) shall be fully open before any attempt is made to move the carriage assemblies.

5.1.5 The carriages will be visually verified to be fully out of the transfer tubes on the Spent Fuel Pool side before FH-V-1A and B are closed.

5.1.6 If the fuel transfer system is used for transfer of anything other than fuel assemblies or control rod assemblies, the operators should be particularly attentive to ensure that uneven weight distribution does not prevent the basket from lowering to the full down and latched position.

5.1.7 For the cable drive system, do not upend the carriage basket unless the carriage and the upender are properly aligned and both the reactor and fuel storage cables are attached to the carriage and tensioned.

5.1.8 For manual operation of the cable drive system, do not operate the winches by hand unless power to the winches is off and the brakes on both winches are released when the reactor and pool cables are attached to the carriage.

5.1.9 The Carriage Jog control is not to be used for carriage movement because the winches are not synchronized in this mode. Carriage Jog should only be used for maintenance to pay out or take up cable when initially installing or replacing winch cables.

5.2 Administrative

5.2.1 Operators shall not leave fuel assemblies unattended in the transfer systems unless relieved by a qualified operator.

5.2.2 Irradiated fuel movements shall not be permitted in the Fuel Handling Building unless an ESF Filtration System is in operation per 1104-15D. (Ref. 1505-1)

6.0 PREREQUISITES

6.1 The following procedures have been completed:

a. M-80, Fuel Transfer System Relief Valve Setting

b. M-81D, Fuel Transfer System Inspection

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.4	
	Importance Rating	2.5	3.1

(K&A Statement) Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Proposed Question: Common 70

Plant conditions:

- A large break LOCA including fuel damage has occurred.
- Both trains of LPI are in service.
- Decay Heat Removal Pump DH-P-1A needs oil to be added to ensure continued operability.
- Significant radiation levels are anticipated in the area of the pump.
- Adding oil to the Decay Heat Removal Pump has been classified as "Protecting Valuable Property".

Which ONE of the following is the maximum TEDE dose limit for this operation?

A. 2 Rem.
B. 5 Rem.
C. 10 Rem.
D. 25 Rem.

Proposed Answer: C. 10 Rem.

Explanation (Optional):

A. Plausible since it is the limit without further approval.
B. Plausible since it is the 10 CFR 20 limit and a facility limit with additional levels of approval.
C. Correct. Facility limit for "Protecting Valuable Property".
D. Plausible since it is the limit for life saving/protecting large populations.

Technical Reference(s): RP-AA-203, Page 7, Table 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: __________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.12	
	55.43		

Comments:

4.4.11. **SUBMIT** a written report of the PSE assigned dose to the individuals involved within 30 days of the PSE.

4.4.12. The dose equivalent received from a PSE is always tracked separately from routine occupational exposure.

4.4.13. Once an exposure is authorized as a PSE, it **cannot** later be treated as a routine occupational exposure. It must be recorded as a PSE, and all the unique limitations, reporting, and record keeping requirements for PSEs shall apply.

4.5. Emergency Exposure Limits **(CM-1)**

4.5.1. Emergency exposure in excess of 25 rem TEDE is to be limited to once in a lifetime.

4.5.2. Emergency personnel are to be informed "before the fact" of possible health effects at the anticipated exposure levels.

4.5.3. For the control of personnel exposures under emergency conditions, **LIMIT** an individual's dose equivalent per activity as follows:

TABLE 2 – EMERGENCY EXPOSURE LIMITS (REM)

TEDE	LDE	SDE	TODE	ACTIVITY
10	30	100	100	Protecting Valuable Property
25	75	250	250	Lifesaving or Protection of Large Populations
> 25	> 75	>250	> 250	Lifesaving or Protection of Large Populations to Workers Fully Aware of the Risks Involved

4.5.4. Emergency exposures shall be voluntary on the part of the involved individual.

4.5.5. **CONSULT** the Emergency Plan Implementing Procedures regarding approval to exceed NRC exposure limits.

5. DOCUMENTATION

5.1. **RETAIN** completed exposure authorizations, including Attachments 1, 2, and 3, in accordance with the station records management program. This records program will include appropriate controls for storage and preservation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.10	
	Importance Rating	2.9	3.3

(K&A Statement) Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Proposed Question: Common 71

Plant conditions:

- 100% power.
- RM-L-1, "RC LETDOWN MONITOR", rapidly progressed to an ALERT alarm status and has now actuated on HI alarm.

Which one of the following is the required action?

A. Place Control Tower on pressurized recirculation.
B. Ensure RC Sample Valves CA-V-1, 2, 3 or 13 closed on interlock.
C. Monitor RM-G-26 and 27, MAIN STEAM LINE Monitors, for rising counts.
D. Announce on Page: "High Activity has been detected in letdown. All unnecessary personnel in the controlled area report to H.P. Checkpoint."

Proposed Answer: D. Announce on Page: "High Activity has been detected in letdown. All unnecessary personnel in the controlled area report to H.P. Checkpoint."

Explanation (Optional):

A. Plausible for concern of activity outside of containment, incorrect because 1202-11 says to ensure it is NOT on recirc prior to sampling RCS.
B. Plausible because it is an immediate action from 1202-12 but is not tied to RM-L-1.
C. Plausible because it is a manual action but applies to RM-A-5, CONDENSER EXHAUST.
D. Correct – immediate action in 1202-11.

Technical Reference(s): 1202-11, pg. 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

1.0 **SYMPTOMS**

A. Alert Alarm on RC letdown radiation monitor RM-L1.

B. High Radiochemical analysis results > 0.35 μCi/gm Dose Equivalent I-131 **OR** > 100/E μCi/gm.

2.0 **IMMEDIATE ACTION**

A. Automatic Action

1. IF a high alarm is received on RM-L1, MU-V2A and MU-V2B close.

B. Manual Action

1. Announce on Page: "High Activity has been detected in letdown. All unnecessary personnel in the controlled area report to H.P. Checkpoint."

3.0 **FOLLOWUP ACTION**

Objective: Determine activity in RCS, minimize radiation levels in the plant, minimize radioactive exposure to personnel, and determine the cause for high activity.

NOTE
RM-L1 low should also have a high alarm.

_____ 1. **IF** RM-L1 is in high alarm, **THEN COMPARE** its reading to RM-L1 low.

_____ 2. **IF** RM-L1 low is not tracking RM-L1, **THEN** have the Shift Manager **EVALUATE** restoring letdown per OP-TM-211-950.

_____ 3. **IF** RM-L1 low is tracking RM-L1, **THEN** manually **CLOSE** or verify closed MU-V2A and MU-V2B.

_____ 4. **IF** letdown **CANNOT** be restored, **THEN COMMENCE** plant shutdown and cooldown to limit pressurizer level to ≤ 330 inches.

_____ 5. **IF** pressurizer level reaches 380 inches while the reactor is critical, **THEN TRIP** the reactor.

_____ 6. **MONITOR** RM-A5 channel for possible increase.

_____ 7. **IF** RM-A5 alert alarms **AND** OTSG tube leak is suspected, **THEN REFER** to OP-TM-EOP-005.

_____ 8. **ENSURE** control tower ventilation is **NOT** on recirculation prior to sampling RCS **OR TAKE** precautions to limit gas activity released to the Control Building.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G2.3.11	
	Importance Rating	2.7	3.2

(K&A Statement) Ability to control radiation releases.

Proposed Question: Common 72

Plant conditions:

- A release is in progress from Waste Gas Decay Tank WDG-T-1A.
- WDG-T-1B is in service.
- The NLO on rounds reports that indicated pressure in WDG-T-1C has lowered 5 PSIG since the last set of readings.

Which one of the following is the required action?

A. Close WDG-47, Waste Gas Release Control Valve.
B. Verify WDG-V-32, WDG-T-1C Outlet Isolation Valve, is locked closed.
C. Verify RM-A-7, WDG-T Discharge Monitor, has remained below the isolation setpoint.
D. Perform a valve lineup on WDG-T-1C to verify that it is NOT aligned to the vent header.

Proposed Answer: A. Close WDG-47, Waste Gas Release Control Valve.

Explanation (Optional):

A. Correct – procedure specified action since an uncontrolled release may be in progress.
B. Plausible since this is an action for the tank being released.
C. Plausible if the unplanned release is not considered.
D. Plausible since a valve misalignment would affect WDG-T-1C pressure.

Technical Reference(s): 1104-27, 3.17 (Page 53) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:

Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 b.13
55.43 ________

Comments:

3.17 Termination of Waste Gas Decay Tank due to a pressure drop in any other gas tank not being released. – **LEVEL 1**

3.17.1 Prerequisites

_____ 3.17.1.1 A Waste Gas Decay Tank is being released to the atmosphere.

_____ 3.17.1.2 Per Enclosure 2, a Waste Gas Decay Tank has lowering pressure and it is not being released to the atmosphere.

3.17.2 Procedure – **Level 1**

_____ 3.17.2.1 Bleed off air from WDG-V-47

_____ 3.17.2.2 Close WDG-V-47 by depressing the Close PB.

_____ 3.17.2.3 Reset WDG-FR-123 setpoint to "0".

_____ 3.17.2.4 Close and lock closed the applicable manual outlet valve on the tank being released. (N/A the valves not open)

Close and lock closed WDG-V-30 for WDG-T-1A. _____
Close and lock closed WDG-V-31 for WDG-T-1B. _____
Close and lock closed WDG-V-32 for WDG-T-1C. _____

_____ 3.17.2.5 Notify the Control Room to update the locked valve entry.

_____ 3.17.2.6 Purge RM-A-7 per Section 3.11 to reduce counts to approximate level prior to the release.

_____ 3.17.2.7 Record the following information in the appropriate sections of the Waste Gas Release Permit:

- Time/Date the release stopped.
- Total duration of the release in minutes
- RM-A-7G, RM-A-8P, RM-A-8I, RM-A-8G, and FR-123 readings
- Applicable WDG-T-1 pressure when the release was terminated
- Δ pressure between the beginning and end of the release

_____ 3.17.2.8 Remove the "Gas Release in Progress" signs at the following locations.

- In the Control Room on PRF at the RM-A-7 ratemeter.
- On the radwaste panel on the WDG-V-47 air controller.

Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	______
10 CFR Part 55 Content:	55.41	b.10
	55.43	______

Comments:

4.1.7 Performing Parallel Procedures

A. Any other procedure actions should be interrupted to perform Reactor Trip Immediate Manual Actions and the initial Symptom Check.

B. Once Immediate Manual Actions and the initial symptom check 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.39	
	Importance Rating	3.3	3.1

(K&A Statement) Knowledge of the RO's responsibilities in emergency plan implementation.

Proposed Question: Common 74

Plant conditions:

- An OTSG tube rupture has resulted in a reactor trip and ESAS actuation.
- The Shift Emergency Director initially classified the event as an ALERT 1115 hours.
- Local notifications of the ALERT have been made.
- The Shift Emergency Director upgraded the classification to SITE AREA EMERGENCY at 1148.

Which ONE of the following identifies the latest time that the upgrade notification to required local agencies can be made and what action stops the clock?

A. 1203; after completion of the initial roll call.
B. 1203; after all required contacts have acknowledged the message.
C. 1248; after completion of the initial roll call.
D. 1248; after all required contacts have acknowledged the message.

Proposed Answer: A. 1203; after completion of the initial roll call.

Explanation (Optional):

A. Correct – 15 minutes is the time limit and the initial roll call meets the criteria described in the Mid-Atlantic States/Local Notification procedure.
B. Plausible because it indicates the message has been received but does not meet procedure criteria.
C. Plausible because the second part is correct but the time is the limit for NRC notification.
D. Plausible because it balances the choices.

Technical Reference(s): EP-MA-114-100, Page 6 (NOTE prior to Step 4.2.4) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: ____________

Learning Objective: ____________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

4.2 STATE / LOCAL NOTIFICATION FORM TRANSMITTAL

4.2.1 When provided with the completed notification form, the designated communicator shall:

1. **ENSURE** that "*Utility Message No.*" has been assigned using a sequential number.

2. **VERIFY** "*Emergency Director Approval*" signature has been entered on the top of form.

3. **REVIEW** form for completeness and **IDENTIFY** any missing information (incomplete blocks) to:

   - Control Room → Shift Manager (Shift Emergency Director)
   - TSC → TSC Director
   - EOF → EOF Director

4.2.2 **CONFIRM** dial tone on NARS line.

4.2.3 **DIAL** the appropriate CODE (CAN No.) listed for the affected station at the top of Roll Call Box on the State/Local Event Notification Form (form EP-MA-114-100-F-01).

1. **IF** the NARS network fails, **CONTACT** agencies using the alternate telephone numbers denoted below each listed agency on the State/Local Event Notification Form.

NOTE: Completion of the initial Roll Call (contact made via dedicated or commercial line with agencies listed) must be performed within 15 minutes of initial classification, reclassification or PAR change.

4.2.4 **REPEAT** the following message while allowing for agencies to come on line:

"This is the Exelon Nuclear [*Station and Facility originating the call*]. **Please standby for a notification message**."

After approximately 10 to 15 seconds, **READ** the following message:

"This is the Exelon Nuclear [*Station and Facility originating the call*]. **Please standby to receive a notification message and respond as the roll is called."**

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G2.4.13	
	Importance Rating	3.3	3.9

(K&A Statement) Knowledge of crew roles and responsibilities during EOP flowchart use.

Proposed Question: Common 75

Plant conditions:

- An automatic reactor trip has actuated.
- The operating crew is performing EOP-001, REACTOR TRIP.

Which ONE of the following is permitted to be performed by a licensed reactor operator without concurrence from the Control Room Supervisor?

A. Reset the Pressurizer level control setpoint to the post-trip value.
B. Tripping MFW Pumps if OTSG A or B Operating Range level is > 97.5%.
C. An "If at any time (IAAT)" action after the step has been read and the condition is met.
D. Any alarm response procedure action step after the step "ENSURE performance of an Alarm Review" has been read.

Proposed Answer: A. Reset the Pressurizer level control setpoint to the post-trip value.

Explanation (Optional):

A. Correct – similar to OS-24 example with actual values deleted to avoid conflicts with other questions.
B. Plausible by the nature of the statement. However, the RO is expected to get SRO concurrence for any action except those cited in OS-24.
C. Plausible because it is an action step in the procedure and the ARO is permitted to control MFWP speed (without SRO concurrence) to maintain valve DP.
D. Plausible because the RO's are permitted to refer to alarm response procedures.

Technical Reference(s): OS-24, Page 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

4.1.16 In-Plant Procedure Usage

A. EVENT PROCEDURE actions performed outside the Control are performed when directed with the applicable procedure section or sections in hand.

B. Simple tasks may be directed to be performed without a copy of the procedure if the following applies:

- The sequence of the steps is controlled by a Reactor Operator.
- Proper 3 part communications are used.

C. If an operator is requested to perform an EVENT PROCEDURE action before completion of a previously directed EVENT PROCEDURE action, the operator notifies the control room team of the uncompleted action.

D. The operator notifies the control room team when the EVENT PROCEDURE action is complete.

4.2 Actions Not Described in Procedures

4.2.1 Taking Action Outside of Approved EVENT PROCEDURE

NOTE
Operators consider the priority of <u>any</u> action performed with respect to potential for crew distraction (e.g. it is not appropriate to ask CRS for concurrence to "match flags" while he is directing EOP actions)

A. Licensed operators may take action without procedural guidance under the following conditions.

- The existing guidance is incorrect or non-conservative due to current plant or equipment conditions AND the Control Room Supervisor provides concurrence for any action taken.
- To minimize <u>immediate</u> personnel hazard/injury or damage to plant equipment
- Actions should be taken to directly compensate for the failure of automatic systems.
- The following actions may be taken by the URO or ARO following a reactor trip w/o CRS concurrence if the action can be taken without distracting the team from addressing a SYMPTOM:

  I. **RESET** pressurizer level setpoint to 25% (100")

  II. **PLACE** FW Pump control in HAND **and ADJUST** pump speed to maintain FW Valve DP between 60 & 90 psid.

Examination Outline Cross-reference: ~~Level~~

	RO	SRO
Tier #		1
Group #		1
K/A #	AA2.01 (015)	
Importance Rating		3.5

What is the K/A Statement

OK

(K&A Statement) Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Cause of RCP failure

Proposed Question: SRO 76

Plant conditions:

- Plant startup in progress IAW 1102-2, "Plant Startup".
- Reactor Power 79%.
- ICS is in full auto, except ULD.

Event:

- Reactor Coolant Pump RC-P-1C ammeter falls to approximately 25%.
- MAP Alarm F-3-2, "RC LOOP B FLOW LO", illuminates.
- RC-P-1C indication light is illuminated Red.
- Reactor power is reducing.
- MAP Alarm H-1-1, "ICS RUNBACK", is illuminated.
- MAP Alarm H-2-1, "ICS IN TRACK", is illuminated.

tripped on OC. Also with a locked rotor the impeller amps ... What is the Basis for Rx trip Stating is plausible

Which ONE of the following describes the event and the appropriate procedure to enter?

Because ICS in MAN

A. RC-P-1C locked rotor, trip the Reactor and go to EOP-001, "Reactor Trip".
B. RC-P-1C locked rotor, secure RC-P-1C in accordance with ~~OP-TM-226-153~~, "Shutdown RC-P-1C", and verify the runback.
C. RC-P-1C sheared shaft, trip the reactor and go to EOP-001 "Reactor Trip".
D. RC-P-1C sheared shaft, secure RC-P-1C in accordance with ~~OP-TM-226-153~~, "Shutdown RC-P-1C", and verify the runback.

Proposed Answer: D. RC-P-1C sheared shaft, secure RC-P-1C in accordance with OP-TM-226-153, "Shutdown RC-P-1C", and verify the runback.

Explanation (Optional):

A. Plausible if the examinee misdiagnoses the event as a locked rotor on RC-P-1C.
B. Plausible if the examinee misdiagnoses the event as a locked rotor on RC-P-1C, and applies dropped impeller actions.
C. Plausible however plant power maybe reduced under these conditions by the runback.
D. Correct answer: Reduced RCP current and reducing flow are indications of a sheared shaft. The plant will runback due to flow or the tripping of the RCP.

Technical Reference(s): OP-TM-MAP-H0101, ICS RUNBACK (Attach if not previously provided)

OP-TM-MAP-F0302, RC LOOP B FLOW LO

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 b.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	AA2.01 (015)	
	Importance Rating		3.5

(K&A Statement) Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Cause of RCP failure

Proposed Question: SRO 76

Plant conditions:

- Plant startup in progress IAW 1102-2, "Plant Startup".
- Reactor Power 79%.
- ICS is in full auto, except ULD.

Event:

- Reactor Coolant Pump RC-P-1C ammeter falls to approximately 25%.
- MAP Alarm F-3-2, "RC LOOP B FLOW LO", illuminates.
- RC-P-1C indication light is illuminated Red.
- Reactor power is reducing.
- MAP Alarm H-1-1, "ICS RUNBACK", is illuminated.
- MAP Alarm H-2-1, "ICS IN TRACK", is illuminated.

Which ONE of the following describes the event and the appropriate procedure to enter?

A. RC-P-1C locked rotor, trip the Reactor and go to EOP-001, "Reactor Trip".
B. RC-P-1C locked rotor, secure RC-P-1C in accordance with OP-TM-226-153, "Shutdown RC-P-1C", and verify the runback.
C. RC-P-1C sheared shaft, trip the reactor and go to EOP-001, "Reactor Trip".
D. RC-P-1C sheared shaft, secure RC-P-1C in accordance with OP-TM-226-153, "Shutdown RC-P-1C", and verify the runback.

Proposed Answer: D. RC-P-1C sheared shaft, secure RC-P-1C in accordance with OP-TM-226-153, "Shutdown RC-P-1C", and verify the runback.

Explanation (Optional):

A. Plausible if the examinee misdiagnoses the event as a locked rotor on RC-P-1C.
B. Plausible if the examinee misdiagnoses the event as a locked rotor on RC-P-1C, and applies dropped impeller actions.
C. Plausible however plant power maybe reduced under these conditions by the runback.
D. Correct answer: Reduced RCP current and reducing flow are indications of a sheared shaft. The plant will runback due to flow or the tripping of the RCP.

Technical Reference(s): OP-TM-MAP-H0101, ICS RUNBACK (Attach if not previously provided)

OP-TM-MAP-F0302, RC LOOP B FLOW LO

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________

55.43 b.5

Comments:

1.0 SETPOINTS

- Loss of one RCP and > 665 MWe
- Loss of two RCP and > 405 MWe
- Loss of one main feed pump and > 560 MWe
- Asymmetric rod fault and > 455 MWe.
- RC flow limit: variable
  MWe equivalent to percent core power $\geq \frac{105 \times \text{Actual RC flow}}{140 \times 10^6 \text{ lbs/hr}}$

(at ~135 x 10^6 lbs/hr, RC flow starts to limit power at 100%)

2.0 CAUSES

- RCP Trip
- MFP Trip
- Dropped Rod
- RCS Flow Degradation

3.0 AUTOMATIC ACTIONS

NOTE: Actual reactor power may vary due to plant efficiencies.

NOTE: ULD STAR module will trip to Manual following runback.

- Unit runs back to:
  - 665 MWe (≈ 75% NI power) at a rate of 50%/minute on a loss of 1 RCP.
  - 405 MWe (≈ 50% NI power) at a rate of 50%/minute on a loss of 2 RCPs.
  - 560 MWe (≈ 68% NI power) at a rate of 50%t/minute on a loss of 1 MFP.
  - 455 MWe (≈ 55% NI power) at a rate of 30%/minute on an Asymmetric rod fault.
  - RC flow limited MWe power level at a rate of 20%/minute on RC Flow degradation.

4.0 MANUAL ACTIONS REQUIRED

- **ENSURE** NI power is reduced to below the limit for the runback condition.
  - **If** ICS manual control is required, **then INITIATE** OP-TM-621-471, ICS Manual Control.
  - **IF** there is a dropped rod, **then GO TO** OP-TM-AOP-062, Inoperable Rod.
- **INITIATE** 1102-4 for power reduction
- **DOCUMENT** required entries in transient cycle logbook.

RC LOOP B FLOW LO

Level 2 – Reference Use

1.0 SETPOINTS

- 62.2 x 10^6 lbs/Hr – RC-14B-FI (CC) (from RC14B-FS)

2.0 CAUSES

- RC-P-1C and / or RC-P-1D malfunction or trip

3.0 AUTOMATIC ACTIONS

- ICS Runback (Ref H-1-1)
- OTSG B BTU Limit (Ref H-1-3)
- Neutron X-Limit to FW (Ref H-1-4)
- Possible Reactor Trip (Ref G-1-1)
- RC-12 TAS (CC) - T_{AVE} control and indication will shift to Loop A

4.0 MANUAL ACTIONS REQUIRED

- **ENSURE** ICS runback and feedwater flow re-ratio.
- **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).

NOTE: Indications of normal RCP operation: amps ≈ 85% (CC) @ 579°F T_{AVE} (80 to 90%), and 560 to 600 amps locally at breaker.

- **OBSERVE** the following **and DETERMINE** faulty RC Pump:
  - RC-P-1C and RC-P-1D Ammeters (CC)
  - Bentley-Nevada – for elevated vibrations (PLF)
- **If** faulty RC Pump is still in operation,
  then PERFORM the following:
  - **START** at least one RC-P-2 pump (Oil Lift)(CC).
  - **START** at least one RC-P-3 pump (Backstop Oil)(CC).
  - **PERFORM** OP-TM-226-150 series procedure to place affected RCP in the Standby mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	G2.1.32 (027)	
	Importance Rating		3.8

(K&A Statement) Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO 77

Plant conditions:

- PORV Block Valve RC-V-2 is closed with power removed, due to the PORV RC-RV-2 failing open.
- 1102-11, "Plant Cooldown", is in progress.
- RCS temperature is 327°F.

To prevent severe over-pressurization of the RCS...

A. The plant must be in cold shutdown within 30 hours of PORV inoperability.
B. ES selected High Pressure Injection Pumps must be in Pull to Lock.
C. Restore power to the PORV Block Valve and open the PORV Block Valve.
D. MU-V-16A/B/C/D are closed with their breakers open, and MU-V-217 IS CLOSED.

Proposed Answer: D. MU-V-16A/B/C/D are closed with their breakers open, and MU-V-217 IS CLOSED.

Explanation (Optional):

A. Plausible since this would be a requirement for an inoperable PORV Block Valve; however the PORV is the inoperable component in this situation and the block valve is allowed to be closed.
B. Plausible since the High Pressure Injection Pumps do have to be off; however the Tech Spec requirement is for the breakers to be racked out.
C. Plausible since this would satisfy the LTOP TS; however it would depressurize the plant.
D. Correct answer. TS 3.1.12.3. With the PORV out of service <329 °F this will prevent overpressurization of the RCS.

Technical Reference(s): TS 3.1.12.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	b.2	

Comments:

3.1.12 Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.1 LTOP Protection

If the reactor vessel head is installed and indicated RCS temperature is ≤ 329°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed, and

b. Pressurizer level is maintained ≤ 100 inches. If pressurizer level is > 100 inches, restore level to ≤ 100 inches within 1 hour.

3.1.12.2 The PORV settings shall be as follows:

a. Low Temperature Overpressure Protection Setpoint

1. When indicated RCS temperature is ≤ 329°F, the LTOP system shall be operable as defined in Specification 3.1.12.1 and

2. The PORV will have a maximum lift setpoint of 552 psig.

With the PORV setpoint above the maximum value, within 8 hours either:

1. restore the setpoint below the maximum value, or

2. verify pressurizer level is ≤ 100 inches indicated and satisfy the requirements of Technical Specification 3.1.12.3 allowing the PORV to be taken out of service.

b. Unless the Low Temperature Overpressure Protection Setpoint is in effect, the PORV lift setpoint will be a minimum of 2425 psig.

With the PORV setpoint below the minimum value, within 8 hours either:

1. restore the setpoint above the minimum value, or

2. close the associated block valve, or

3. close the PORV, and remove power from PORV

4. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	G2.4.50 (029)	
	Importance Rating		3.3

(K&A Statement) Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: SRO 78

Plant conditions:

- Reactor power is 65% following the removal of RC-P-1A from service due to high vibration.
- Turbine load is 498 MWe.
- Both Main Feedwater Pumps are in Hand.
- All other equipment lineups are normal.

Event:

- MAP Alarm F-1-1 "RCP Motor Trip", illuminates.
- MAP Alarm F-3-2 "RC Loop B Flow Lo", illuminates.
- MAP Alarm H-3-1 "ICS Runback", illuminates.
- MAP Alarm H-2-1 "ICS In Track", illuminates.
- B Loop RCS flow is reducing.
- RC-P-1D Breaker status Green light illuminates and Red light extinguishes.
- RC-P-1D amber Disagreement light illuminates.
- ICS Runback commences to reduce plant load.

Seems too easy LOD = 1 RO for SRO ready question. Only knowledge to answer you need to know on feedwater 76% pwr with only 2 RCPs RPS setpoint trip. Suggest a more difficult question.

Which ONE of the following actions is required?

A. Trip the reactor in accordance with OP-TM-EOP-001, "Reactor Trip".
B. Verify the plant runs back to <405 MWe in accordance with OP-TM-MAP-H0101, "ICS Runback".
C. Verify Feedwater flow re-ratios to both OTSGs in accordance with OP-TM-MAP-F0101, "RCP Motor Trip".
D. Manually reduce Feedwater flow using the Feedwater Pumps in accordance with OP-TM-401-472 and 473, "Manual Control of FW-P-1A" and "FW-P-1B", respectively.

Proposed Answer: A. Trip the reactor in accordance with OP-TM-EOP-001, "Reactor Trip".

Explanation (Optional):

A. Correct answer. 65% power is above the two Reactor Coolant Pump RPS trip setpoint.
B. Plausible since this is the MWe setpoint for the ICS runback; however Reactor power is above the RPS setpoint.
C. Plausible since feedwater flow will re-ratio in this situation to equalize flows in both OTSGs; however Reactor power is above the RPS trip setpoint.
D. Plausible since the Main Feedwater Pumps are in Hand; however Reactor Power is above the RPS trip setpoint.

Technical Reference(s): OP-TM-EOP-001, Reactor Trip (Page 1) (Attach if not previously provided)
TS Table 2.3-1 RPS Trip Setting Limits

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	G2.4.50 (029)	
	Importance Rating		3.3

(K&A Statement) Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: SRO 78

Plant conditions:

- Reactor power is 65% following the removal of RC-P-1A from service due to high vibration.
- Turbine load is 498 MWe.
- Both Main Feedwater Pumps are in Hand.
- All other equipment lineups are normal.

Event:

- MAP Alarm F-1-1 “RCP Motor Trip”, illuminates.
- MAP Alarm F-3-2 “RC Loop B Flow Lo”, illuminates.
- MAP Alarm H-3-1 “ICS Runback”, illuminates.
- MAP Alarm H-2-1 “ICS In Track”, illuminates.
- B Loop RCS flow is reducing.
- RC-P-1D Breaker status Green light illuminates and Red light extinguishes.
- RC-P-1D amber Disagreement light illuminates.
- ICS Runback commences to reduce plant load.

Which ONE of the following actions is required?

A. Trip the reactor in accordance with OP-TM-EOP-001, “Reactor Trip”.
B. Verify the plant runs back to <405 MWe in accordance with OP-TM-MAP-H0101, “ICS Runback”.
C. Verify Feedwater flow re-ratios to both OTSGs in accordance with OP-TM-MAP-F0101, “RCP Motor Trip”.
D. Manually reduce Feedwater flow using the Feedwater Pumps in accordance with OP-TM-401-472 and 473, “Manual Control of FW-P-1A” and “FW-P-1B”, respectively.

Proposed Answer: A. Trip the reactor in accordance with OP-TM-EOP-001, “Reactor Trip”.

Explanation (Optional):

A. Correct answer. 65% power is above the two Reactor Coolant Pump RPS trip setpoint.
B. Plausible since this is the MWe setpoint for the ICS runback; however Reactor power is above the RPS setpoint.
C. Plausible since feedwater flow will re-ratio in this situation to equalize flows in both OTSGs; however Reactor power is above the RPS trip setpoint.
D. Plausible since the Main Feedwater Pumps are in Hand; however Reactor Power is above the RPS trip setpoint.

Technical Reference(s): OP-TM-EOP-001, Reactor Trip (Page 1) (Attach if not previously provided)

TS Table 2.3-1 RPS Trip Setting Limits

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 b.5

Comments:

TABLE 2.[illegible]

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).

REACTOR TRIP

1.0 ENTRY CONDITIONS

- Any unplanned condition requiring an automatic or manual trip signal, (OS-24 Attachment A)
- A symptom of core cooling upset occurs while shutdown prior to DHR operation. (OS-24 Attachment D)

2.0 IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 2.1 **PRESS** both Reactor Trip **and** DSS pushbuttons.	
___ 2.2 **VERIFY** REACTOR SHUTDOWN.	___ 1. **TRIP** 1L-02 **and** 1G-02. ___ 2. **If** the Reactor is shut down, **then GO TO** Step 2.3. ___ 3. **If** Main FW is **not** available, **then**: ___ **ENSURE** Main Turbine is tripped. ___ **ENSURE** EFW is actuated. ___ 4. **MAINTAIN** primary-to-secondary heat transfer. ___ 5. **When** RCS pressure < 2500 psig, **then INITIATE** HPI, IAW OP-TM-211-901, "Emergency Injection HPI/LPI. ___ 6. **When** the Reactor is shutdown, **then CONTINUE**.
2.3 **PRESS** Turbine Trip PB.	
2.4 **VERIFY** the Turbine stop valves are CLOSDED.	___ **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).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	EA2.15 (038)	
	Importance Rating		4.4

(K&A Statement) Ability to determine or interpret the following as they apply to a SGTR: Pressure at which to maintain RCS during S/G cooldown

Proposed Question: SRO 79

Plant conditions:

- Reactor is shutdown due to Steam Generator tube leak in OTSG 1A.
- All four RCPs are operating.
- OTSG 1A is isolated at a level of 87%.
- OTSG 1B level is 25 inches.
- OTSG 1A pressure 875 psig.
- OTSG 1B pressure 727 psig.
- BWST level is 54 ft.
- No offsite dose increases have been noted by Radiological Assessment.
- RCS Thot is 460°F.
- RCS Tcold 455°F.
- RCS Pressure is 965 psig.
- RCS cooldown rate is 105°F/hr.

Which ONE of the following actions must be taken?

A. Shutdown one Reactor Coolant Pump per loop in accordance with 1102-11, "Plant Cooldown".
B. Minimize subcooling margin in accordance with OP-TM-EOP-010, Rule 6, "Pressurized Thermal Shock (PTS)".
C. Reduce Subcooling margin to 30 °F by opening the PORV in accordance with OP-TM-EOP-010, Guide 8 "RCS Pressure Control".
D. Preferentially steam OTSG 1A to maintain level on scale in accordance with OP-TM-EOP-005, "OTSG Tube Leakage".

Proposed Answer: B. Minimize subcooling margin in accordance with OP-TM-EOP-010, Rule 6, "Pressurized Thermal Shock (PTS)".

Explanation (Optional):

A. Plausible as 1102-11 will be performed along with EOP-005; however 1102-11 secures RC-P-1C and RC-P-1D.
B. Correct answer. With RCS temperature <525 °F and cooldown rate >100°F/hr SCM is to be minimized.
C. Plausible since this is and action to be taken if OTSG pressure is approaching or is >1000 psig; however Guide 8 does not use the PORV only the spray valve.
D. Plausible since this is an action that would be taken prior to isolating the OTSG.

Technical Reference(s): OP-TM-EOP-005, OTSG Tube Leakage, (Page 5) (Attach if not previously provided)
OP-TM-EOP-010, Guide 12, RCS Stabilization Following OTSG Isolation
OP-TM-EOP-010, Rule 6 Pressurized Thermal Shock (PTS)
1102-11, Plant Cooldown (Page 8)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

PTS

6

Rule 6
Pressurized Thermal Shock (PTS)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** one of the following: - RCS T_{AVG} > 525°F - RCS cooldown rate < 100°F/HR	**MINIMIZE** SCM IAW Guide 8.
2. **VERIFY** one of the following: - MU-V-16A, B, C, D **and** MU-V-217 are CLOSED - At least <u>one</u> RCP operating	**MINIMIZE** SCM IAW Guide 8.
3. **VERIFY** one of the following: - RCS T_{AVG} > 255°F - RCS cooldown rate < 30°F/HR	**MINIMIZE** SCM IAW Guide 8.

3.2.2 Plant Cooldown

> **NOTE**
>
> If conditions required by Enclosure 2 are not satisfied or any time as directed by the CRS, HOLD RCS pressure and temperature stable, using the pressure control and heat removal methods applicable for plant conditions.

3.2.2.1 To ensure all control rods are fully inserted:

____ 1) **ENSURE** turbine bypass valve control is in HAND.

> **NOTE**
>
> Apply reactivity monitoring requirements IAW Enclosure 5.

____ 2) **If** Control Rod Group 8 is withdrawn **and** CRD breakers are closed, **then**

____ **SELECT** Group 8 (Diamond Group Select Switch)

____ **INSERT** control rod Group 8 to 0%

____ **SELECT** OFF (Diamond Group Select Switch)

____ 3) **ENSURE** RPS is actuated (at least two channels tripped). Use OP-TM-641-421 to TRIP RPS channels.

3.2.2.2 Shutdown RC pumps as follows:

____ a) **Monitor** RCS cooldown rate **and ADJUST** TBV position as necessary when RCPs are shutdown.

____ b) **SHUTDOWN** RC-P-1C IAW OP-TM-226-153

____ c) **SHUTDOWN** RC-P-1D IAW OP-TM-226-154

> **NOTE**
>
> The “initial target” RCS pressure is 600 psig and RCS temperature is 350°F, except as follows:
>
> If OP-TM-216-201 (202) are <u>not</u> to be performed, then the “initial target” RCS pressure is 500 psig and RCS temperature is 350°F.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	AA2.06 (054)	
	Importance Rating		4.3

(K&A Statement) Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): AFW adjustments needed to maintain proper T-ave. and S/G level

Proposed Question: SRO 80

Plant conditions:

- Reactor tripped due to a loss of offsite power (LOOP).
- Natural Circulation flow has been verified.
- The Atmospheric Dump Valves are in automatic.
- Tave is 543 °F and lowering.
- RCS Cooldown rate is 62 °F/hr.
- OTSG 1A level 57%.
- OTSG 1B level 61%.
- OTSG 1A Pressure 905 psig and lowering.
- OTSG 1B Pressure 900 psig and lowering.
- All three Emergency Feedwater Pumps are running.
- EFW Flow to OTSG 1A is 422 gpm.
- EFW Flow to OTSG 1B is 477 gpm.

Which ONE of the following actions must be taken?

A. Initiate OP-TM-EOP-010, Rule 3, "Excessive Primary to Secondary Heat Transfer", to isolate B OTSG.
B. Close Emergency Feedwater Isolation Valves (EF-V-52s) in accordance with EOP-010, Guide 16.3, "EFW Failure", to isolate the Emergency Feedwater Control Valves (EF-V-30s).
C. Close the Emergency Feedwater Control Valves in accordance with OP-TM-EOP-010, Guide 15, "EFW Actuation Response", to stop the OTSG pressure reduction.
D. Take manual control of the Atmospheric Dump Valves and close them in accordance with OP-TM-411-451, "Manual Control of TBVs/ADVs", to lower the cooldown rate.

Proposed Answer: C. Close the Emergency Feedwater Control Valves in accordance with OP-TM-EOP-010, Guide 15 "EFW Actuation Response", to stop the OTSG pressure reduction.

Explanation (Optional):

A. Plausible if the examinee misdiagnoses an overcooling event; however Tave is not <540°F.
B. Plausible since this is an action to take if the EF-V-30s are failed open; however placing the valves in hand first to try and close them would be the expected action.
C. Correct answer. OTSGs are above minimum level and pressure is >100 psig below desired.
D. Plausible since this is an action that would be taken if the ADVs were open; however the examinee should diagnose the valves as closed since OTSG pressure is below setpoint and the valves are in automatic.

Technical Reference(s): OP-TM-EOP-010, Guide 15, EFW Actuation Response (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

Guide 15
EFW Actuation Response

IAAT EFW is actuation is required, **then**:

____ 1. **ENSURE** EF-P-1, EF-P-2A, **and** EF-P-2B start.

____ 2. **DISPATCH** an Auxiliary Operator (AO) to EF-V-30 area.

____ 3. **If** EFW pump disch. pressure < OTSG pressure, **then INITIATE** Guide 16.

☐ 4. **IAAT** OTSG pressure is more than 100 psig below desired, **then THROTTLE** EFW flow if permitted by Rule 4

☐ 5. **IAAT** only EF-P-2A **or** EF-P-2B is operating, **then THROTTLE** EF-V-30s to maintain total EFW flow < 515 GPM.

____ 6. **If** EFW is being manually initiated **and** the OTSG is DRY, **then VERIFY** Guide 13, DRY OTSG requirements are satisfied.

____ 7. **ENSURE** EF-V-30A/D and EF-V-30B/C control OTSG level at setpoint (Rule 4).

____ 8. **If** Shift Management concurrence is obtained, **then** EFW flow may be controlled using one **or** both EF-V-30 valves for each OTSG.

☐ 9. **IAAT** an EFW failure occurs, **then INITIATE** Guide 16.

☐ 10. **IAAT** EF-P-1 operation is required,

 ____ 1 **If** OTSG Tube leakage symptoms exist **and** aux steam is available, **then OPEN** AS-V-4.

 ____ 2 **If** OTSG A **and** B pressure < 200 psig, **then**,

  ____ **CLOSE** the breaker for MS-V-10A [1C DC switch #1]

  ____ **CLOSE** the breaker for MS-V-10B [1D DC switch #1]

  ____ **Jog OPEN** MS-V-10A **or** B to maintain EF-P-1 speed > 3300 RPM.

 ____ 3 **If** OTSG A **and** B pressure < 150 psig **and** aux steam is available, **then OPEN** AS-V-4.

____ 11. **ENSURE** AH-E-24A **or** AH-E-24B is operating.

____ 12. **When** CO-T-1A **or** CO-T-1B < 5 ft, **then INITIATE** Guide 17.

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. **VERIFY** SCM > 25°F.	**If** both OTSGs are available **and** OTSG tube leak < 1 gpm, **then FEED** with EFW > 215 gpm / OTSG. **If** only one OTSG is available **or** OTSG tube leak > 1 gpm, **then FEED** with EFW > 430 gpm to the good OTSG.
2. **VERIFY** an RCP is operating **or** incore temperature is stable or lowering.	**FEED** OTSG at maximum available EFW flow.
3. **VERIFY** EFW is available.	**If** SCM < 25°F, **or** [all RCPs OFF **and** incore temp is rising], **then FEED** with MFW at > 1.0 Mlbm/hr.
4. There is no minimum required flow rate.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	G2.1.32 (065)	
	Importance Rating		3.8

(K&A Statement) Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO 81

Plant conditions:

- Reactor was manually tripped due to a loss of Instrument Air.
- Secondary Instrument Air pressure is currently 45 psig.
- Primary Instrument Air pressure is currently 51 psig.
- Intermediate Closed Cooling Water Flow is 265 gpm.
- RCP Seal Injection flow is 30 gpm.

MU-V-20, RCP Seal Injection Isolation Valve, is required to be blocked. . . .

A. CLOSED to ensure maximum HPI flow in the event of an automatic or manual ES actuation.
B. CLOSED to prevent the valve from opening and thereby inadvertently shocking the RCP seals when Instrument Air Pressure is restored.
C. OPEN to ensure maximum available makeup flow with the normal makeup flowpath isolated.
D. OPEN to ensure continued SI flow, with an operator assigned and available to close if containment isolation is required.

Proposed Answer: D. Blocked OPEN to ensure continued SI flow, with an operator assigned and available to close if containment isolation is required.

Explanation (Optional):

A. Plausible since this is the reason for closing Charging Header isolation MU-V-18 on an ES actuation.
B. Plausible since the reason is a basis for isolating seals after SI has been lost.
C. Plausible since the valve is blocked OPEN but the reason is incorrect.
D. Correct answer.

Technical Reference(s): OP-TM-AOP-028, Loss of Instrument Air, Page 1 (Attach if not previously provided)
Op-TM-211-000, Page 23/49
TS 3.6 Basis 3-41c

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.2

Comments:

LOSS OF INSTRUMENT AIR

1.0 ENTRY CONDITIONS

Any of the following conditions **and** DHR is **not** in service:

- Primary IA header pressure < 80 psig (PI-222) and lowering
- Secondary IA header pressure < 80 psig (PI-1403) and lowering
- Report of a major air leak (observed air leak and IA pressure lowering)

2.0 IMMEDIATE ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 2.1 **ANNOUNCE** loss of instrument air over plant page and radio. (AO response per OS-24 Attachment E)	

NOTE

Once a containment isolation valve is blocked open, an operator (with ability to communicate with the control room) must remain in the aux bldg to close the valve if containment isolation is required.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2.2 **IAAT** IA pressure is < 60 psig (PI-222 or PI-1403), **then PERFORM** the following: ___ A. **If** SI Flow > 22 gpm, **then DISPATCH** an operator to block Open MU-V-20 (AB 305: behind MU-F-4A/B), ___ B. **If** ICCW Flow > 550 gpm, **then DISPATCH** an operator to block Open IC-V-4 (AB 305: W of RB wall) **and** IC-V-3 (AB 281:A shielded area) ___ C. **ENSURE** reactor is tripped. ___ D. **GO TO** Section 4.0.	

BASES

The Reactor Coolant System conditions of COLD SHUTDOWN assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures. The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the Reactor Coolant System is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn. The Reactor Building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

The primary Containment Isolation Valves (CIVs) are identified in UFSAR Table 5.3-2. Additional vent, drain, test and other manually operated valves which complete the containment boundary are identified in the containment integrity checklist. For the purpose of this specification, check valves and relief valves identified in the containment integrity checklist are defined to be active valves.

The loss of redundant capability for containment isolation is limited for all penetrations after which the containment penetration must be isolated. Isolation of certain penetrations may require the closure of multiple CIVs due to piping branches.

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.

2. For those CIVs where the second barrier is a closed system within the Reactor Building, there is no other CIV to isolate the penetration. If operability cannot be regained, the valve must be closed within 72 hours or the plant must commence shut down. An action time of 72 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a containment isolation boundary and the relative importance of supporting containment integrity.

The definition of Containment Integrity permits normally closed CIVs, except for the 48 inch purge valves, to be unisolated intermittently or manual control to be substituted for automatic control under administrative control. Administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment (Reference 1). The dedicated individual can be responsible for closing more than one valve provided that the valves are in close vicinity and can be closed in a timely manner. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the containment penetrations containing these valves may not be opened under administrative control.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. Analysis has demonstrated that a purge isolation valve is capable

ATTACHMENT 7.1
MU (211) System Mode Definition
Page 7 of 32

Device	Description	Critical or Locked	ES Standby Mode	LTOP Mode	Shutdown Mode
MU-V-16A	HPI Valve A	Critical	Closed	Closed	Closed
MU-V-16A-BK	1A ES Valves MCC Unit 4B	Critical	On	Off	Off
MU-V-16B	HPI Valve B	Critical	Closed	Closed	Closed
MU-V-16B-BK	1A ES Valves MCC Unit 4C	Critical	On	Off	Off
MU-V-16C	HPI Valve C	Critical	Closed	Closed	Closed
MU-V-16C-BK	1B ES Valves MCC Unit 4B	Critical	On	Off	Off
MU-V-16D	HPI Valve D	Critical	Closed	Closed	Closed
MU-V-16D-BK	1B ES valves MCC Unit 4C	Critical	On	Off	Off
MU-V-17-EX1	MU-V-17 H/A Station (0-100% Dmd) (RC1-LIC)	N/A	Auto	Auto	Hand / 0% Demand
MU-V-17-POS	MU-V-17 VALVE POSITIONER (HAND/AUTO)	N/A	Auto	Auto	Hand
MU-V-18	RCS makeup isolation valve (CIV)	Critical	Open	Open	Closed
MU-V-18 HW	Handwheel for MU-V-18	Critical / Locked	Handwheel is Full Closed/Fully Backed Out	Handwheel is Full Closed/Fully Backed Out	Handwheel is Full Closed/Fully Backed Out
MU-V-18-Test Switch	MU-V-18 ES Test Switch (PCR)	N/A	Normal	Normal	Normal
MU-V-20	RCP Seal Injection isolation valve (CIV)	Critical	Open	Open	Closed
MU-V-20 HW	Handwheel for MU-V-20	Critical / Locked	Handwheel in Full Closed /Fully Backed Out	Handwheel in Full Closed /Fully Backed Out	Handwheel in Full Closed /Fully Backed Out
MU-V-25	RCP Seal Return Isolation Valve (CIV)	Critical	Open	Open	Closed
MU-V-25-BK	1A ES Valves MCC Unit 4D	Critical	On	On	On

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	G2.4.30 (037)	
	Importance Rating		3.6

(K&A Statement) Emergency Procedures / Plan: Knowledge of which events related to system operations/status should be reported to outside agencies.

Proposed Question: SRO 82

Which ONE of the following events would require four (4) hour report to the NRC Operations Center, in accordance with LS-AA-1020, "Reportability Reference Manual"?

A. Activation of ERDS, following declaration of an ALERT.
B. Initiating a plant shutdown for an OTSG tube leak of 0.5 gpm.
C. Completion of a cooldown to replace a leaking Code Safety Valve.
D. Cold shutdown loss of offsite power (LOOP), both diesel load onto buses.

Proposed Answer: B. Initiating a plant shutdown for an OTSG tube leak of 0.5 gpm.

Explanation (Optional):

A. Plausible activation of ERDS requires notification, time however is 1 hour.
B. An OTSG tube leak >0.1 gpm is a Condition of License in technical Specifications requiring Shutdown, per F-aa of LS-AA-1020 10CFR50.72(b)(2)(i) would apply.
C. Plausible however cooldown does not require notification unless T.S. initiated and then would be 8 hour.
D. Plausible would be an E-plan entry unusual event, would require 1 hour notification.

Technical Reference(s): LS-AA-1020 Page 4 F-aa (Attach if not previously provided)
T.S. Amendment to license (8)

Proposed references to be provided to applicants during examination: LS-AA-1020, Reportability Reference Manual

Learning Objective: ________ (As available)

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

Question History:	Last NRC Exam	____
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	____
10 CFR Part 55 Content:	55.41	____
	55.43	b.5

Comments:

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

(8) <u>Repaired Steam Generators</u>

In order to confirm the leak-tight integrity of the Reactor Coolant System, including the steam generators, operation of the facility shall be in accordance with the following:

1. Prior to initial criticality, the licensee shall submit to NRC the results of the steam generator hot test program and a summary of its management review.

2. The licensee shall confirm baseline primary-to-secondary leakage rate established during the steam generator hot test program. If leakage exceeds the baseline leakage rate by more than 0.1 gpm*, the facility shall be shut down and leak tested. If any increased leakage above baseline is due to defects in the tube free span, the leaking tube(s) shall be removed from service. The baseline leakage shall be re-established, provided that the leakage limit of Technical Specification 3.1.6.3 is not exceeded.

3. The licensee shall complete its post-critical test program at each power range (0-5%, 5%-50%, 50%-100%) in conformance with the program described in Topical Report 008, Rev. 3, and shall have available the results of that test program and a summary of its management review, prior to ascension from each power range and prior to normal power operation.

4. The licensee shall conduct eddy-current examinations, consistent with the extended inservice inspection plan defined in Table 3.3-1 of NUREG-1019, either 90 calendar days after reaching full power, or 120 calendar days after exceeding 50% power operation, whichever comes first. In the event of plant operation for an extended period at less than 50% power, the licensee shall provide an assessment at the end of 180 days of operation at power levels between 5% and 50%, such assessment to contain recommendations and supporting information as to the necessity of a special eddy-current testing (ECT) shutdown before the end of the refueling cycle. (The NRC staff will evaluate that assessment and determine the time of the next eddy-current examination, consistent with the other provisions of the license conditions.) In the absence of such an assessment, a special ECT shutdown shall take place before an additional 30 days of operation at power above 5%.

*If leakage exceeds the baseline leakage rate by more than 0.1 gpm during the remainder of the Cycle 8 operation, the facility shall be shutdown and leak tested. Operation at leakage rates of up to 0.2 gpm above the baseline leakage rate shall be acceptable during the remainder of Cycle 8 operation. After the 9R refueling outage, the leakage limit and accompanying shutdown requirements revert to 0.1 gpm above the baseline leakage rate.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	AA2.04 (060)	
	Importance Rating		3.4

(K&A Statement) Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: The effects on the power plant of isolating a given radioactive gas leak

Proposed Question: SRO 83

Plant conditions:

- Reactor power is 100%.
- Normal equipment lineups exist.
- Waste Gas Decay Tank 1B is in service at 65 psig.
- Waste Gas Decay Tank 1A is at 0 psig.
- Waste Gas Decay Tank 1C is at 78 psig awaiting release paperwork.

Event:

- A small leak has been discovered on the Waste Gas Delay Tank relief valve.
- It has been determined that the Waste Gas Delay Tank must be isolated to replace the relief valve.

Given these conditions the CRS must ....

A. initiate 1102-4, "Power Operations", to place the unit in Hot Shutdown to allow isolation of the vent header.
B. initiate 1104-27, "Waste Disposal – Gaseous", to isolate and bypass the Waste Gas Delay Tank to allow continued operation of the Waste Gas System.
C. initiate 1104-27, "Waste Disposal – Gaseous", to shut down the Waste Gas System and isolate all WDL tanks on the header to terminate the leak.
D. initiate 6610-ADM-4250.11, "Waste Gas Release Permit for Waste Gas Decay Tanks 1B and 1C", to allow depressurization of the Waste Gas System for repairs.

Proposed Answer: B. initiate1104-27 "Waste Disposal – Gaseous" to bypass the Waste Gas Delay Tank to allow continued operation of the Waste Gas System.

Explanation (Optional):

A. Plausible since a shutdown, cooldown and depressurization would be required if the vent header had to be isolated for repairs.
B. Correct answer. The Delay tank can be bypassed for repairs and have the waste gas system remain in service.
C. Plausible since there is a normal system shutdown procedure but the RCS must be shutdown and depressurized to do so.
D. Plausible if the examinee believes the waste gas tanks must be depressurized to isolate the system for repairs.

Technical Reference(s): 1104-27, Waste Disposal – Gaseous (Page 15) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

_____ 4. De-energize the control power to the Waste Gas System by opening switches inside the LWDS Panel: Can be N/A with Control Room Supervisor concurrence.

No. 1 in the 1B-RWD 120 Volt Distribution Panel.

No. 9 in the 1B-RWD 120 Volt Distribution Panel.

CAUTION

Switch No. 1 also supplies power to process Group A, B, C, and miscellaneous wastes - unit selection of the L.W.D. system. Prior to de-energizing switch No. 1, be sure that no process mentioned above is in progress. If so, wait until the process is finished.

5. Removal of the Waste Gas Delay Tank from service during System Operation. Can be N/A with Control Room Supervisor concurrence.

_____ a. Open the Waste Gas Delay Tank bypass valve - WDG-V-11.

_____ b. Close the inlet, WDG-V-9, and the outlet, WDG-V-10, to the Waste Gas Delay Tank.

_____ c. Close the drain pot vents, WDG-V-15, and WDG-V-13.

_____ d. Open the drain pot drain, WDG-V-14. The Waste Gas Delay Tank is now isolated from the System.

_____ e. Check as closed WDG-V-66.

6. Removal of the Waste Gas Decay Tank from service during system operation. Can be N/A with Control Room Supervisor concurrence.

_____ a. If the Waste Gas Compressors are discharging to the tank selected for removal, manually shift the waste gas compressor discharge to another tank per Section 3.5.

_____ b. Remove the three amp control fuse from the inlet valve control located in the Liquid Waste Control panel labeled:

_____	WDG-V-24	FU-V-24G
_____	WDG-V-26	FU-V-26G
_____	WDG-V-28	FU-V-28G

_____ c. Close/verify close the reuse valve from the Waste Gas Decay Tank that is selected for isolation, WDG-V-79, WDG-V-83, or WDG-V-87.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	G2.2.25 (067)	
	Importance Rating		3.7

(K&A Statement) Equipment Control Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Proposed Question: SRO 84

If the Control Room must be evacuated due to a fire, Tech Specs requires the Remote Shutdown System to contain enough control and instrumentation functions to ...

A. place and maintain the unit in Hot Standby.
B. place and maintain the unit in Hot Shutdown.
C. cooldown the unit to the Cold Shutdown condition.
D. cooldown the unit to the Refueling Shutdown condition.

Proposed Answer: B. place and maintain the unit in Hot Shutdown.

Explanation (Optional):

A. Plausible if the examinee does not know the definition of Hot Standby.
B. Correct answer per TS Basis.
C. Plausible since the unit can cooldown; however TS basis does not require this capability.
D. Plausible since the unit can cooldown; however TS basis does not require this capability.

Technical Reference(s): TS 3.5.7 Basis (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ____
55.43 b.2

Comments:

3.5.7 REMOTE SHUTDOWN SYSTEM

Applicability

Applies to the operability requirements for the Remote Shutdown System Panel "B" Functions in Table 3.5-4 during STARTUP, POWER OPERATION AND HOT STANDBY.

Objectives

To assure operability of the instrumentation and controls necessary to place and maintain the unit in HOT SHUTDOWN from a location other than the control room.

Specification

The minimum number of functions identified in Table 3.5-4 shall be OPERABLE. With the number of functions less than the minimum required, restore the required function to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within an additional 12 hours.

Bases

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from locations other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as HOT SHUTDOWN.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in HOT SHUTDOWN. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches HOT SHUTDOWN following a unit shutdown and can be maintained safely in HOT SHUTDOWN for an extended period of time.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	EA2.2 (E09)	
	Importance Rating		4.0

(K&A Statement) Ability to determine and interpret the following as they apply to the (Natural Circulation Cooldown): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: SRO 85

Plant conditions:

- The reactor has tripped due to a loss of offsite power (LOOP).
- Cooldown is in progress using the Atmospheric Dump Valves.
- Neither OTSG has tube leakage.

Which ONE of the choices fills in the blank of the following statement?

As a minimum, RCS pressure must be maintained above the ______ line of EOP-010, Figure 1, "RCS PRESSURE-TEMPERATURE LIMITS".

A. SAT CURVE
B. 25 °F SCM
C. PREVENT HEAD BUBBLE C
D. 250 °F SCM

Proposed Answer: C. PREVENT HEAD BUBBLE

Explanation (Optional):
A., B., D. Plausible because each choice is an "as identified" line on the curve.
C. Correct. With no tube leakage and adequate heat transfer, maintain pressure above that line during an NC cooldown.

Technical Reference(s): OP-TM-EOP-010, Guide 10, Natural Circulation (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	b.5	

Comments:

Guide 10
Natural Circulation

IAAT all RCPs are off, **then**:

A. **If** all of the following conditions exist, **then** adequate Natural Circulation exists:

- RCS T_{HOT} minus T_{COLD} stabilizes at less than 50°F.
- T_{HOT} < 600°F.
- Incore temperature stabilizes **and** tracks T_{HOT}.
- Cold leg temperatures approach saturation temperature for secondary side pressure.
- OTSG heat removal is indicated by feeding **or** steaming with stable OTSG pressure.
- SCM ≥ 25°F.

____ TIME Natural Circulation was VERIFIED

B. **If** OTSG tube leakage < 1 GPM and Primary to Secondary Heat Transfer exists, **then**

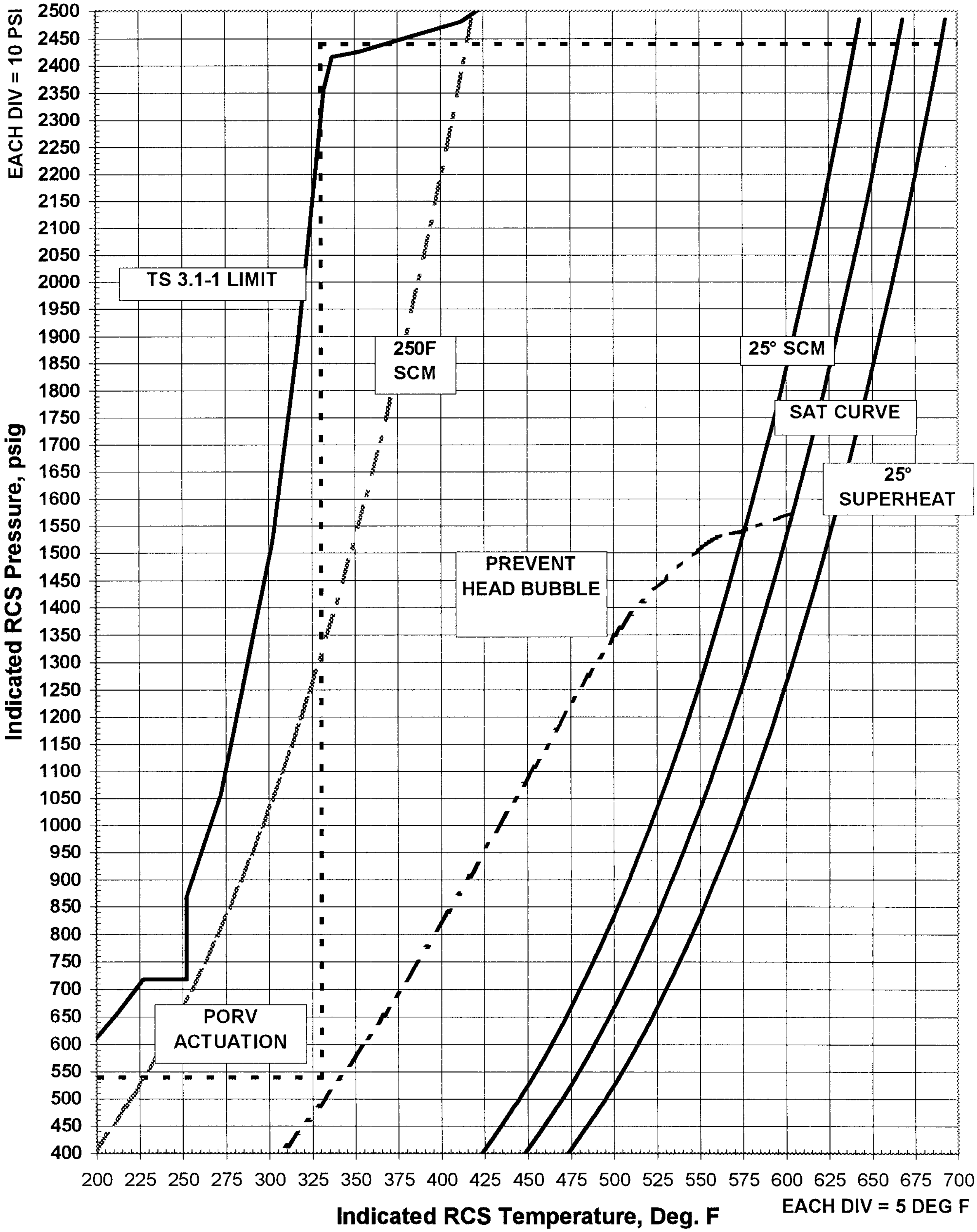
MAINTAIN RCS pressure above the "PREVENT RV HEAD BUBBLE" curve on Figure 1 to avoid developing a steam bubble in the Reactor Vessel head.

2. **If** one of the following conditions exists:

   - SCM < 25°F in hot leg of an isolated OTSG
   - T_{HOT} on an isolated OTSG is more than 50°F above active loop T_{HOT}.

   then FEED OTSG with EFW (or MFW if EFW is not available) until T_{HOT} A and B are within 20°F to minimize the potential for hot leg voids,
   or REDUCE RCS cooldown rate.

FIGURE 1
RCS PRESSURE-TEMPERATURE LIMITS



Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	A2.02 (010)	
Importance Rating		3.9

(K&A Statement) Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Spray valve failures.

Proposed Question: SRO 86

Plant conditions:

- With the plant at 100% power Main Feedwater Pump FW-P-1A trips.
- During the transient RCS Pressure peaks at 2245 psig and the 1B 4160 V bus trips.

Current plant conditions:

- Reactor power has stabilized at 65%.
- RCS pressure is 2030 psig and lowering slowly.
- All available pressurizer heaters are energized.
- RC Drain Tank pressure is <2 psig.
- RC Drain Tank Temperature is stable.

Which ONE of the following actions must be taken FIRST?

A. Transfer Group 9 Pressurizer Heaters to 1P 480V Bus in accordance with OP-TM-220-901, "Emergency Power Supply For Pressurizer Heaters".
B. Trip the Reactor due to lowering RCS pressure to prevent an automatic trip and go to OP-TM-EOP-001, "Reactor Trip".
C. Close the Pressurizer Spray Block Valve RC-V-3 to prevent a reactor trip in accordance with OP-TM-AOP-043, "Loss of Pressurizer (Solid Ops Cooldown)".
D. Close the PORV Block Valve RC-V-2 to prevent a reactor trip in accordance with OP-TM-MAP-G0308, "RC Press Narrow RNG HI/LO".

Proposed Answer: C. Close the Pressurizer Spray Block Valve RC-V-3 to prevent a reactor trip in accordance with OP-TM-AOP-043, "Loss of Pressurizer (Solid Ops Cooldown)".

Explanation (Optional):

A. Plausible since power has been lost to these heaters but this operation is no permitted with the reactor critical.
B. Plausible since this is an action to be taken in OP-TM-MAP-G0308 before an automatic trip occurs; however closing the Spray Block Valve would occur first.
C. Correct answer. Indications of Pressurizer Spray Valve failure. AOP-043, Step 3.5.
D. Plausible since closing the PORV Block Valve is an action to take if the PORV is open or leaking; however the RC Drain tank indication provided to not signify a leaking or open PORV.

Technical Reference(s): AOP-043, Step 3.5, Page 3 (Attach if not previously provided)
OP-TM-MAP-G0308

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

Level 2 – Reference Use

1.0 SETPOINTS

Hi - >2255 psig or Lo <2055 psig from:

RC3-PR Hot Leg A Narrow Range Channel 1

RC3 PIS Hot Leg B Narrow Range Channel

2.0 CAUSES

- RC-V-1 Pressurizer spray valve Open
- Faulty pressurizer heater operation
- RCS temperature rising on Lowering
- Primary coolant leak
- RC-RV-2 PORV Open
- Failure of controlling RCS Pressure Instrument RC3-PR or RC3-PIS (CC)
- Loss of ICS-Auto or Hand Power

3.0 AUTOMATIC ACTIONS

- Reactor Trip at RCS pressure >2355 psig or <1900 psig.

4.0 MANUAL ACTIONS REQUIRED

4.1 **If** RCS pressure is HI,
then PERFORM the following:

4.1.1 **ENSURE** pressurizer heaters are Off.

4.1.2 Fully **OPEN** RC-V-1 PZR Spray Control Valve.

4.1.3 **RETURN** RC-V-1 PZR Spray Control Valve to automatic.

4.2 **If** RCS pressure is Lo,
then PERFORM the following:

4.2.1 **If** RC-RV-2 PORV is Open **and** RCS pressure <2400 psig,
then CLOSE RC-V-2.

4.2.2 **ENSURE CLOSED** RC-V-1 PZR Spray Control Valve.

4.2.3 **If** failure of RC-V-1 is suspected,
then CLOSE RC-V-3 Pressurizer Spray Line Isol Valve as required.

4.2.4 **IAAT** Spray Line ΔT approaches 250°F, **then CYCLE** RC-V-3.

4.2.5 **If** pressurizer level is $\geq$ 80 inches,
then ENSURE pressurizer heaters are energized.

4.2.6 **If** RC-V-1 and RC-V-3 are Open, **then**

1. **RESET** Thermal Overload for RC-V-1 1A ES MCC Unit 9B.
2. **ATTEMPT** to Close RC-V-1.
3. Prior to RPS actuation in low pressure,

    A. **TRIP** the Reactor.

    B. **TRIP** All Reactor Coolant Pumps.

4.2.7 **If** pressurizer heater capacity is inadequate to maintain RCS pressure,
then GO TO OP-TM-AOP-043, "Loss Of Pressurizer".

4.2.8 **If** RC-V-3 is Closed,
then PLACE an EST on RC-V-3.

3.0 FOLLOW-UP ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3.1 **IAAT** <u>all</u> of the following conditions exist: – Valid pressurizer level indication is available, – Pressurizer level < 370", – Adequate pressurizer heaters are available to control RCS pressure, **then GO TO** Section 6.0, "Return To Normal".	
___ 3.2 **VERIFY** at least one valid pressurizer level indication.	___ **GO TO** Section 4.0 "Loss Of Pressurizer Level Indication".
☐ 3.3 **IAAT** pressurizer level > 315" (PZR Level Hi Hi), **then INITIATE** Plant Shutdown using 1102-4, "Power Operation" and 1102-10, "Plant Shutdown".	
☐ 3.4 **IAAT** pressurizer level > 370", **or** Tave < 350 °F with Pzr > 100", **then GO TO** Section 5.0 "Solid Plant Operation".	
___ 3.5 **CLOSE** RC-V-3, "Pressurizer Spray Block Valve".	
___ 3.6 **VERIFY** PORV is <u>**not**</u> leaking.	___ **CLOSE** RC-V-2, "PORV Block Valve".

CAUTION

Excessive SCM may challenge PORV and Code Safety valves.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3.7 **IAAT** SCM < 40 °F, **then PERFORM** the following: ___ 1 **PLACE** MU-V-17 in hand. ___ 2 **RAISE** Makeup flow to maintain >40 °F SCM.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	G2.1.30 (012)	
	Importance Rating		3.4

(K&A Statement) Conduct of Operations: Ability to locate and operate components, including local controls.

Proposed Question: SRO 87

Plant conditions:

- The Reactor is at 100%.
- Reactor Protection System "A" is in MANUAL BYPASS.
- Reactor Protection Channel "B" is being bypassed in accordance with OP-TM-641-455, "RPS CHANNEL MANUAL BYPASS", for maintenance activities.

Which ONE of the choices completes the following statement?

The CRS must verify no other RPS cabinet is currently tripped at the __________ and then has "A" RPS Cabinet tripped by, removing from bypass and placing RPS "A" __________.

A. top of outside RPS "A"; Contact Monitor Test Module to "TEST OPERATE"
B. "A" Reactor Trip Module Sub Assembly; Contact Monitor Test Module to "TEST OPERATE"
C. top of outside RPS "A"; two sub assembly "TRIP TEST" switches to Trip
D. "A" Reactor Trip Module Sub Assembly; two sub assembly "TRIP TEST" switches to Trip

Proposed Answer: A. top of outside RPS "A"; contact Monitor Test Module to "TEST OPERATE"

Explanation (Optional):

A. Correct – top outside lights show other cabinet status. Test operate is per OP-TM-641-455.
B. Plausible because it is the correct switch but wrong location to verify other cabinets.
C. Plausible it is the correct location to verify other cabinets but incorrect method to trip cabinet. This would only trip associated CRD breaker; cabinet would remain reset.
D. Plausible because it balances the choices.

Technical Reference(s): OP-TM-641-455, Page 2 (Attach if not previously provided)
OP-TM-641-421, Page 2

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

4.0 MAIN BODY

> **WARNING**
>
> Performing next step/substeps will lead to a Reactor Trip if intent is to de-energize another RPS channel.

4.1 **If** another Channel in Manual Bypass due to an equipment problem, **then** perform the following:

1. **VERIFY** **no** other RPS channel is tripped.
2. Place Channel already in Manual Bypass to a tripped state IAW OP-TM-641-421, Tripping and Resetting RPS Channels.
   A. **VERIFY** Manual BY-PASS lamp Dim on Reactor Trip Module. (**CM-1**)
   B. **VERIFY** alarm MAP G-3-1 Clear.
3. **VERIFY** **no** Vital Bus work in progress that could affect operable RPS Channels.

NOTE: Following step will place RPS in a 1 out of 2 Logic for a reactor trip.

4. **PLACE** selected RPS Channel Manual Bypass Switch to BYPASS position using Key #6.
   – **VERIFY** Manual By-Pass lamp Bright on Reactor Trip Module. (**CM-1**)
   – **VERIFY** MANUAL BY-PASS lamp Bright on outside of RPS cabinet. (**CM-1**)
   – **VERIFY** alarm MAP G-3-1 In.
5. **DOCUMENT** Manual Bypass position changes in CR Logbook.

4.0 MAIN BODY

NOTE: This section trips RPS channels. Resetting RPS channels is accomplished in section 5.0 of this procedure.

4.1 **VERIFY** Shift Management concurrence to trip RPS channel(s). ____

1. **If** all RPS channels will be tripped to support plant cooldown, **then GO TO** step 4.2. ____

2. **MARK** columns in step 4.2 as **N/A** for the channels **not** to be tripped. ____

3. **VERIFY** all other RPS channels are Reset (**not** tripped). ____

CAUTION

Placing more than one RPS channel in test will cause a Reactor Trip.

4.2 **PERFORM** the following at the identified RPS Cabinet (s):

Action	RPS Cabinet Initials			
	A	B	C	D
ENSURE channel is **not** in Manual Bypass				
PLACE Contact Monitor Test module switch to TEST OPERATE.				
VERIFY Alarm MAP G-1-2 RPS Channel Trip In.				
VERIFY Reactor Trip module (RTM) TEST TRIP lamp Bright.				
VERIFY respective RTM Protective Subsystem lamp Bright.	No. 1	No. 2	No. 3	No. 4
PLACE Contact Monitor Test module switch to OPERATE.				

4.2.1 **LOG** tripped RPS channel(s) in Control Room logbook. ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	G2.1.32 (026)	
	Importance Rating		3.8

(K&A Statement) Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO 88

Plant conditions:

- An RCS LOCA results in reactor trip from 100% power.
- OP-TM-EOP-010, "Rule 1 - Loss of Subcooled Margin" was completed.
- ESAS equipment has operated except Decay Heat Pump1B, which tripped.
- OP-TM-EOP-006, "LOCA Cooldown" has been initiated.
- Decay Heat Removal crossconnect valves DH-V-38A and DH-V-38B are open.
- Decay Heat Removal Pressurizer Spray Isolation Valve RC-V-4 is closed.
- Guide 22 RB Sump Recirc has been initiated and Sump Recirc Valves DH-V-6A and DH-V-6B are open.
- Decay Heat Removal Train A flow is 1415 gpm.
- Decay Heat Removal Train B flow is 1355 gpm.
- Reactor Building Spray Pump 1A Flow is 1450 gpm.
- Reactor Building Spray Pump 1B Flow is 1100 gpm.
- Reactor Building Sump Level is 90 inches.

Which ONE of the following is the proper action:

A. Throttle DH-P-1A flow to <2700 total gpm to prevent pump runout in accordance with OP-TM-EOP-010, "Guide 22 – RB Sump Recirculation".
B. Raise DH-P-1A flow to ≤3300 gpm to maximize reactor vessel cooling in accordance with OP-TM-EOP-010, "Rule 2 - HPI-HPI/LPI Throttling".
C. Throttle Reactor Building Spray Pump 1A flow to < 1400 gpm to ensure pump NPSH in accordance with OP-TM-214-000, "Building Spray System".
D. Raise Reactor Building Spray flow in both trains to ≤1700 gpm to maximize spray in accordance with OP-TM-214-901, "RB Spray Operation".

Proposed Answer: C. Throttle Reactor Building Spray Pump 1A flow to < 1400 gpm to ensure pump NPSH in accordance with OP-TM-214-000, "Building Spray System".

Explanation (Optional):

A. Plausible since 2700 gpm would be the limit if RC-V-4 was open.
B. Plausible since 3300 gpm would be the limit if NOT on sump recirc.
C. Correct Answer – BS-P-1A flow exceeds procedure limitation of 1400 gpm when from the RB Sump.
D. Plausible since 1700 gpm is the limit if RB Spray Pumps are taking suction from the BWST; however the flow orifice should limit flow to 1100 gpm.

Technical Reference(s): OP-TM-214-000, Building Spray System (Step 2.2.1) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
55.41 ______
55.43 b.5

Comments:

2.1.2 BS-T-2 Sodium Hydroxide Tank (NaOH Tank) & BS-T-1 (old Sodium Thio Tank)

- To avoid tank damage from potential over pressure or vacuum the tank vent valves BS-V-12A (BS-T-1) & BS-V-12B (BS-T-2) shall be maintained Open whenever fill or drain is in progress.
- Pumping a cold solution into the warm tank vapor space could cause a rapid depressurization of the tank and threaten tank integrity. Manway should be fully Open or consult with engineering should temperature of fluid being added be < 60°F (BS-T-2) or < 45°F (BS-T-1).

2.1.3 Admin

- Inadvertent operation of the Building Spray System is highly undesirable. Avoid unnecessary introduction of spray into Reactor Building.
- Do **not** simultaneously operate the DHR system (DHR mode) and the Reactor Building Spray System unless required for ESAS actuation. This will prevent inadvertent transfer of reactor coolant to the BWST.

2.2 Limitations

2.2.1 Limit BS pump flow to < 1700 gpm while taking suction from BWST. Limit BS pump flow to < 1400 gpm when taking suction from the RB sump to ensure adequate pump NPSH.

2.2.2 To avoid accidental spraying of Reactor Building, with RCS pressure less than 500 psig during a normal plant cooldown, BS-V-1A and BS-V-1B will be Closed (Reactor Building Spray Discharge Isolation Valves), and the breakers for these valves will be Open.

3.0 REFERENCES

3.1 Building Spray Procedures

3.1.1 OP-TM-214-101, ES Standby Alignment for BS System

3.1.2 OP-TM-214-151, Reactor Building Spray for Shutdown

3.1.3 OP-TM-214-201, IST of BS-P-1A and Valves

3.1.4 OP-TM-214-202, IST of BS-P-1B and Valves

3.1.5 OP-TM-214-203, IST of ECCS Bypass - BS Valves

3.1.6 OP-TM-214-204, Comprehensive Pump Test of BS-P-1A and Valves

3.1.7 OP-TM-214-205, Comprehensive Pump Test of BS-P-1B and Valves

3.1.8 OP-TM-214-251, BS and DH Floor Drain Inspection

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	A2.12 (062)	
	Importance Rating		3.6

(K&A Statement) Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Restoration of power to a system with a fault on it.

Proposed Question: SRO 89

PLANT CONDITIONS:

- Reactor power 50% following a loss of Feedwater Pump 1B.
- VA-P-1B main vacuum pump is OOS for maintenance.
- CO-P-1C condensate pump is OOS for maintenance.
- CO-P-1A & CO-P-1B condensate pumps are operating.
- CO-P-2A & CO-P-2C condensate booster pumps are operating.

Event:

- 1A 4160 Volt Bus trips.

Which ONE of the following represents:

(1) The potential impact on the plant.
(2) The mitigation strategy.

OK / STET

A. (1) The reactor will trip due to loss of feedwater.
(2) Initiate Emergency Feedwater i~~n accordance with OP-TM-EOP-010, Guide~~ 15, "Emergency Feedwater Actuation Response".

B. (1) Condensate Booster Pump CO-P-2B will auto-start.
(2) Reduce power to within the capability of the Condensate/Booster pumps ~~in~~ accordance with OP-TM-AOP-010, ~~"Loss of 1A 4160 Volt Bus~~".

C. (1) Overheating of the A Main Transformer due to loss of cooling.
(2) Reduce load ~~in accordance with 1104-2, "Power Operations", to maintain transformer~~ temperatures within specifications.

D. (1) Vacuum will begin to degrade due to loss of power to the operable vacuum pumps.
(2) Provide power to ~~start either operable Main~~ Vacuum Pump ~~in accordance with~~ OP-TM-~~731-901~~, "Energize 1C 480 ~~Volt Bus Using~~ 1N Bus Crosstie".

STET

Proposed Answer: D (1) Vacuum will begin to degrade due to loss of power to the operable vacuum pumps.
(2) Provide power to start either operable Main Vacuum Pump in accordance with OP-TM-731-901, "Energize 1C 480 Volt Bus Using 1N Bus Crosstie. Energize 1C 480 Volt Bus Using 1N Bus Crosstie".

Explanation (Optional):

A. Plausible if the examinee thinks the second feedwater pump will trip based on power and pump combinations; however FW-P-1A will remain on line since there is still one condensate and booster pump running.
B. Plausible since with a different condensate alignment this would occur.; however the standby condensate booster pump will not start since only one condensate pump is running.
C. Plausible since one train of Main Transformer cooling will be lost; however the redundant cooling should prevent overheating.
D. Correct – no power to VA-P-1C because 1A 4160 V cannot be recovered with a fault. Power must be cross-tied from 1N.

Technical Reference(s): OP-TM-AOP-010, Loss of 1A 4160 Volt Bus, (Step 3.16) (Attach if not previously provided)
OP-TM-731-901, Energize 1C 480 Volt Bus Using 1N Bus Crosstie (Note after step 4.2.8)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

4.2.7 **If** 1N bus load @ N1-02 has increased more than 60 amps
or 1N bus load @ N1-02 > 75 amps while reactor is critical, **then**

DE-ENERGIZE additional loads on 1A CW MCC, 1A TP MCC and 1C TP MCC. ____

TRACK changes in 1N bus load. ____

TRACK breakers opened on Attachment 1 ____

4.2.8 **If** EDG is supplying 1D 4160V bus,
then VERIFY EDG load is maintained IAW OP-TM-861-901 or OP-TM-864-901. ____

NOTE: Electrical loads on 1C 480V bus may be energized or shutdown. Monitor 1N bus load to track changes and maintain cognizance of estimated 1N bus cross tie load.

Approximate load (amps at 4160V) for:

Main vacuum pump 1A or 1C	8 amps each
Secondary Closed Pump 1A	10 amps

4.2.9 **If** CB H&V MCC is de-energized,
then MONITOR battery room temperature and declare battery inoperable if room temperature < 70°F. ____

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Both Auxiliary Boilers are unavailable. "B" Auxiliary Boiler electrical power can be restored if 1C Turbine Plant 480V bus is energized.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.16 **INITIATE** OP-TM-731-901, "Energize 1C 480V Bus Using 1N Bus Cross Tie".	
___ 3.17 **ENSURE** AH-P-2B is operating.	
___ 3.18 **VERIFY** GS-E-1B is operating.	___ **INITIATE** OP-TM-314-451, "Swapping Gland Seal Exhausters".
___ 3.19 **VERIFY** AH-E-2B is operating.	___ **INITIATE** OP-TM-823-402, "Swapping Reactor Compartment Ventilation Fans (AH-E-2A/B)".
___ 3.20 **INITIATE** OP-TM-511-489, "FAA Notification Of NDCT Aircraft Warning Light Failure" for "B" cooling tower.	
___ 3.21 **If** the reactor is shutdown, **and** an Auxiliary Boiler is **not** available, **then ADJUST** OTSG pressure to cooldown the RCS as required to maintain OTSG lower downcomer vs. FW differential temperature < 442 °F limit.	
___ 3.22 **NOTIFY** chemistry of loss of sampling and condensate chemical feed ability.	
___ 3.23 **NOTIFY** security and dispatch an operator to check EG-Y-2 operation.	

• • • **END** • • •

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	G2.1.27 (103)	
Importance Rating		2.9

(K&A Statement) Conduct of Operations: Knowledge of system purpose and or function.

Proposed Question: SRO 90

Which ONE of the following conditions exceeds a design assumption for Reactor Building (RB) performance during a design basis accident and the basis for the limit?

A. RB Pressure is 2.2 psig with the unit at 50% power; RB Pressure may exceed design limits.
B. RB Temperature above the 320' Elevation is 125 °F with the unit at 100% power; Temperature assumptions for equipment environmental qualification may be exceeded.
C. OTSG Sample valve CA-V-5A is failed closed and CA-V-4A is stuck open with the unit at 50% power; RB Isolation may not be established when actuated.
D. RB Equipment Hatch inner door is open for seal replacement with the unit at 100% power; The outer door may exceed ΔP limits and fail.

Proposed Answer: A. RB Pressure is 2.2 psig with the unit at 50% power; RB Pressure may exceed design limits.

Explanation (Optional):
A. Correct – this exceeds the TS limit and is the basis.
B. Plausible since this exceeds the TS limit for below the 320' Elevation.
C. Plausible since these are containment isolation valves but only one must close or be closed.
D. Plausible since hatch doors are TS required but either one is capable of holding design pressure and it is permissible to have one open at a time.

Technical Reference(s): TS 3.6.4 (Page 3-41) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	b.2	

Comments:

3.6 REACTOR BUILDING

Applicability

Applies to the CONTAINMENT INTEGRITY of the reactor building as specified below.

Objective

To assure CONTAINMENT INTEGRITY.

Specification

3.6.1 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (Section 1.7) shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200 degrees F or greater.

c. Nuclear fuel is in the core.

3.6.2 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% $\Delta k/k$ shall not be made by control rod motion or boron dilution unless CONTAINMENT INTEGRITY is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual Containment Isolation Valves (CIVs) which should be closed are closed and are conspicuously marked.

3.6.6 When CONTAINMENT INTEGRITY is required, if a CIV (other than a purge valve) is determined to be inoperable:

a. For lines isolable by two or more CIVs, the CIV(s)* required to isolate the penetration shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, at least one CIV* in the line will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

b. For lines isolable by one CIV, where the other barrier is a closed system, the line shall be isolated by at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 72 hours or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

* All CIVs required to isolate the penetration.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	G2.1.32 (011)	
Importance Rating		3.8

(K&A Statement) Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: SRO 91

Plant conditions:

- Reactor power is 100%.
- Loss of ICS Auto power occurred 20 minutes ago.
- Normal equipment lineups existed prior to the event.
- OP-TM-AOP-027, "Loss of ATA or ICS Auto Power", has been initiated.

As The Control Room Supervisor you must direct the CRO to...

A. trip the reactor in accordance with OP-TM-AOP-043, "Loss of Pressurizer (Solid Ops Cooldown)", when Pressurizer Level rises above 315 inches to prevent taking the plant solid.
B. commence a normal plant shutdown in accordance with OP 1102-4 "Power Operations", to prevent exceeding 315 inches in the Pressurizer while critical.
C. take manual control of Normal Makeup Valve MU-V-17 in accordance with OP-TM-211-472, "Manual Pressurizer Level Control", to maintain level between 80 and 385 inches.
D. reduce Seal Injection flow to 22 gpm in accordance with OP-TM-211-476, "Seal Injection Control – Console Operations", to prevent the need for a plant shutdown or a reactor trip.

Proposed Answer: C. take manual control of Normal Makeup Valve MU-V-17 in accordance with OP-TM-211-472, "Manual Pressurizer Level Control", to maintain level between 80 and 385 inches.

Explanation (Optional):

A. Plausible since a reactor trip would be required by OP-TM-AOP-043 at >370 inches.
B. Plausible since a plant shutdown would have to be commenced if level is >315 inches; however it is not a criticality limit.
C. Correct answer – Pressurizer level will be rising slowly without operator intervention.
D. Plausible since reducing seal injection flow will slow the rise in Pressurizer level; however the procedure says to maintain seal injection flow 32-40 gpm with any RCPs running.

Technical Reference(s): OP-TM-211-472, Manual Pressurizer Level Control, 3.1.1 (Attach if not previously provided)

(Page 1)
OP-TM-AOP-027, Loss of ATA
or ICS Auto Power (Page 5)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

MANUAL PRESSURIZER LEVEL CONTROL

1.0 PURPOSE

1.1 Manual control of Pressurizer level using MU-V-17, 217 or 92. This procedure is used when automatic control is **not** functioning or if manual control is preferable for a specific plant evolution.

2.0 MATERIAL AND SPECIAL EQUIPMENT - None

3.0 PRECAUTIONS, LIMITATIONS, AND PREREQUISITES

3.1 Precautions

3.1.1 With the reactor critical pressurizer level must be maintained between 80 and 385" (TS 3.1.3)

3.1.2 With Tavg < 329°F, the RV head installed and a MU pump breaker racked in, pressurizer level must be maintained less than 100 inches (TS 3.1.12)

3.1.3 To maintain pressurizer heater operation, pressurizer level must be maintained > 80 inches. Heaters will automatically de-energize at < 80".

3.2 Limitations

3.2.1 If Rx power > 20% then maintain pressurizer level between 200″ to 240″.

3.2.2 If the Rx is critical and less than 20% power, then maintain pressurizer level within the band of OP-TM-211-472 attachment 7.2.

3.2.3 If the Reactor is shutdown and RCS > 329°F, then maintain pressurizer level between 80″ to 120″.

3.2.4 If the Reactor is shutdown and RCS < 329°F, then maintain pressurizer level between 80″ to 100″.

3.3 Prerequisites

3.3.1 **VERIFY** MU is in LTOP or ES Standby Mode IAW OP-TM-211-000, Makeup and Purification System. ____

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3.8 **INITIATE** OP-TM-401-472 "Manual Control of FW-P-1A".	
___ 3.9 **INITIATE** OP-TM-401-473 "Manual Control of FW-P-1B"	
___ 3.10 **If** MU-V-17 is in HAND, **then INITIATE** OP-TM-211-472 "Manual Pressurizer Level Control".	
___ 3.11 **INITIATE** OP-TM-211-950 "Restoration Of Letdown Flow".	
___ 3.12 **If** MU-V-32 is in HAND, **then INITIATE** OP-TM-211-476 "Seal Injection Control – MU-V-32 Console Operations".	
___ 3.13 **PLACE** pressurizer level LO LO interlock switch to BYPASS. (Key #2 inside ICS/NNI Power Monitoring Cabinet, Key #214)	
☐ 3.14 **IAAT** pressurizer level < 80" on RC-LI-777A, **then PLACE** pressurizer heaters in OFF: ___ Bank 1 ___ Bank 2 ___ Bank 3 ___ Bank 4 ___ Bank 5	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	A2.05 (071)	
	Importance Rating		2.6

(K&A Statement) Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power failure to the ARM and PRM systems.

Proposed Question: SRO 92

Plant conditions:

- 100% power.
- An approved release is in progress from Waste Gas Decay Tank (WDG-T-1A).

Event:

- Power is lost to RM-A-7, Gaseous Waste Discharge Tank Monitor.

Which one of the following describes the correct action and the subsequent release of WDG-T-1A?

A. Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1, "Rad Mon System Trouble". Release may resume as long as RM-A-8 is operable.
B. Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1, "Rad Mon System Trouble". RM-A-7 must be restored to operable before resuming the release.
C. WDG-V-47 remains open. Continue the release in accordance with 1104-27, "Waste Disposal – Gaseous", as long as RM-A-8 is operable.
D. WDG-V-47 remains open. Manually terminate the release in accordance with 1104-27, "Waste Disposal – Gaseous" until RM-A-7 is restored to operable.

Proposed Answer: A. Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1," Rad Mon System Trouble". Release may resume as long as RM-A-8 is operable.

Explanation (Optional): Explanation (Optional):

A. Correct – Power failure actuates the associated interlock. 1104-27 was recently revised to allow release with RM-A-7 and/or RM-A-8.
B. Plausible because the first part is correct but the release can be resumed with only RM-A-8 operable.
C. Plausible because the second part is correct.
D. Plausible because the choices are balanced.

Technical Reference(s):	MAP Alarm Response C-3-1 Rad Mon System Trouble	(Attach if not previously provided)
	Drawing 209-707	
	1104-27, 3.7.1.3 (Page 22)	

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

ALARM:

RAD MON SYSTEM TROUBLE

SET POINTS:

Loss of voltage to any rad monitor or indicator pointer below zero
Failure mode alarm from digital ratemeter

CAUSES:

Loss of voltage to the following rad monitor detector or amplifiers:

1. RM-G-1 thru RM-G-7, RM-G-9 thru RM-G-21 or RM-G-24 thru RM-G-27
2. RM-L-1 thru RM-L-7 or RM-L-9
3. RM-A-1 thru RM-A-9 or RM-A-15

The following failure modes for a digital ratemeter
The 946A-200-M24 digital ratemeter can experience the following failure modes for which alarms are provided:

- Low Signal Failure
  A low Signal Failure will be detected if the pulses reported to the ratemeter by the preamplifier are zero for five minutes. This may indicate a failure in the electrometer circuit of the detector and preamplifier. An error code of E0011 will be displayed on the ratemeter. However, the display of the ratemeter may read zero for five minutes or more without a low signal failure if the preamplifier is reporting a dose rate less than the under range setpoint value.
- Auto Zero Failure
  An Auto Zero Failure will occur if the preamplifier zero offset correction is near it's limit. An error code E0012 will be displayed on the ratemeter. This failure indicates the auto zero adjustment on the electrometer board in the preamplifier is necessary.
- High Voltage Failure
  The ion chamber is biased at 500 volts from the preamplifier and produces an output current proportional to the radiation absorbed in the chamber. The preamplifier checks the ion chamber integrity every four minutes by pulsing the high voltage and checking the response. Failure of this check indicates a loss of continuity between the preamplifier and detector or bad connections. This High Voltage Failure will indicate an E0011 error code on the display of the ratemeter. This failure may also occur if the high voltage drops to a level such that the ion chamber cannot achieve 100% collection at full scale.
- Loop Failure (communications)
  A Loop Failure is received if the ratemeter does not receive a valid message from the preamplifier within 10 seconds. This failure is indicative of bad seating of the serial communications board, bad connections in the preamplifier, bad connections at the P2 connector of the ratemeter, etc. An error code of E0007 will be displayed on the ratemeter.

_____ 3. Close gas recycle valve WDG-V-25 (27 or 29), using the pushbuttons on the L.W.D. panel in the Auxiliary Building.

_____ 4. Vent the remaining 10 psig of gas in Waste Gas Decay Tank WDG-T-1A (1B or 1C) if desired to the Station Ventilation Stack per Section 3.7 prior to refilling the Waste Gas Decay Tank. Note that this is only necessary if it is desired to fully depressurize the Waste Gas Tank.

Performed By________________________________ Date____________
Signature

Reviewed By SRO
or RO License________________________________ Date____________
Signature

CAUTION

All releases to the vent stack must be accomplished with an approved waste gas release permit.

NOTE

If a gas release is terminated for any reason prior to being empty, ensure it is re-started within 12 hours if possible to prevent 4 to 6 hours of additional work by Chemistry and Rad Con to re-analyze the tank contents. Per 6610-ADM-4250.11 gas releases may be secured for up to 12 hours without re-sampling for a new gas release permit.

3.7 Waste Gas Decay Tank (WDG-T-1A) Disposal - **Level 1**

3.7.1 Prerequisites

_____ 3.7.1.1 WDG-T-1A is isolated IAW section 3.14

_____ 3.7.1.2 The WCS/CRS/SM verified that WDG-T-1A has had a hold time to ensure compliance with ODCM, Section 2.2.2.

_____ 3.7.1.3 During release of gaseous waste from WDG-T-1A, the following conditions must be met:

_____ 1. **ENSURE** requirements of ODCM Part 1, Section 2.1.2 are satisfied using Waste Gas Discharge monitor, RM-A-7, and/or Auxiliary and Fuel Handling Building Exhaust monitor, RM-A-8.

_____ 2. Waste gas decay tank discharge valve, WDG-V-47, must be operable.

_____ 3. WDG-FR-123 must be operable.

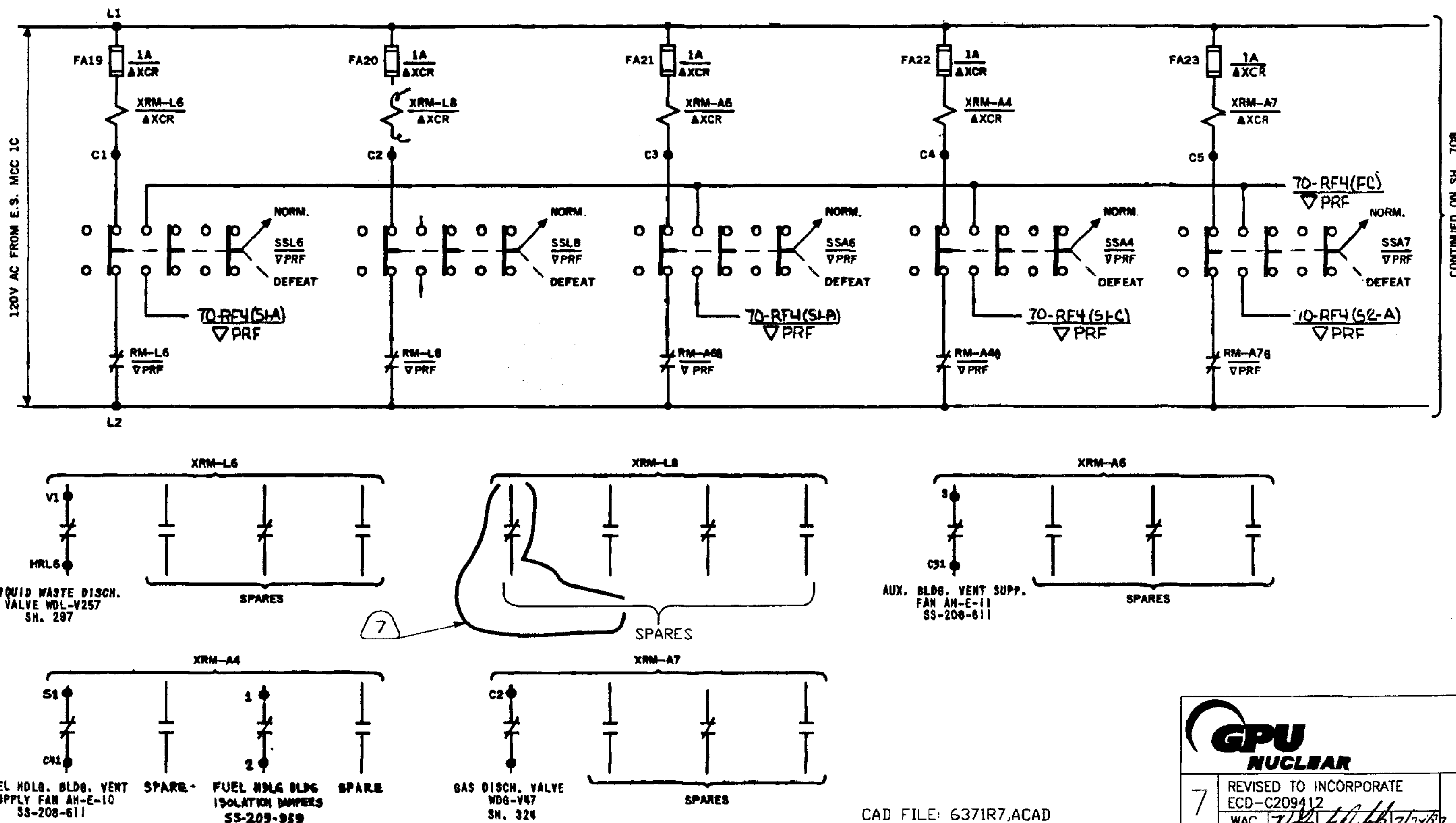
REFERENCE DWGS.:
LEGEND: SS-208-001
SW. DEVELOPMENT: SH. 009
INDEX: SS-209-001

NOTES:
1. ALL XRM RELAYS ARE CLARK 5U4-2
2. RM-RELAY CLOSE ON HIGH RADIATION OR LOSS OF POWER.

METROPOLITAN EDISON COMPANY THREE MILE ISLAND NUCLEAR STATION UNIT 1	MADE RF	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.		
ELECTRICAL ELEMENTARY DIAGRAM	CHK'D. FFB SQ. CF. GC CF. OPN. JCG	4192	SS-209-707	7
D.C. & MISCELLANEOUS	ENG.	WORK ORDER	SIZE DRAWING	REV.
	ISSUE DOC'S. KG - FB -			

REGULATORY REQUIRED

RADIATION MONITORING INTERLOCKS



CAD FILE: 6371R7,ACAD
THIS IS A COMPUTER GENERATED DWG
DO NOT REVISE MANUALLY

GPU NUCLEAR

REV	DRAFT	CHECK	APPROVED	DATE	APP
7	REVISED TO INCORPORATE ECD-C209412 WAC			7/27/87	

REVISION

DATE	RELEASED FOR	ENGR.
	CONSTRUCTION	
	BIDDING PURPOSES ONLY	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	G2.1.23 (086)	
	Importance Rating		4.0

(K&A Statement) Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO 93

Plant conditions:

- Reactor is in Hot Shutdown.
- Normal equipment lineups exist.

Event:

- PLF-1-1 "Reactor BLDG fire", alarm is received.
- Temperatures in the Reactor Building are approximately 98 °F rising slowly.
- An NLO reports the panel fire alarm is for RB-FZ-1A, Reactor Building 281' North Outside "D" ring.

As Control Room Supervisor, which ONE of the following procedural actions will be directed from AOP-001, "FIRE"?

A. Initiate OP-TM-811-901, "Containment Fire Service", to place the containment fire service system in service.
B. Initiate OP 1102-11, "Plant Cooldown", to commence a plant cooldown.
C. Initiate OP-TM-534-901, "RB Emergency Cooling Operations", to start the RB Emergency Cooling System.
D. Initiate OP-MA-001-007, "Fire Protection System Impairment Control", to remove alarm RB-FZ-1A, allowing other zones to actuate an alarm if the fire spreads.

Proposed Answer: A. Initiate OP-TM-811-901, "Containment Fire Service", to place the containment fire service system in service.

Explanation (Optional):

A. Correct – Spool piece not normally installed. AOP-001 directs installation of spool piece per OP-TM-811-901.
B. Plausible since a cooldown may be required; however it is not one of the Fire procedure actions.
C. Plausible since RB Emergency Cooling would have to be initiated for high Reactor Building Temperature; however no RB temperature alarms are presently in and temperature is not excessive.
D. Plausible since no alarm reflash will occur with an alarm actuated but an active, valid alarm would not be bypassed or defeated.

Technical Reference(s): OP-TM-AOP-001, Fire (Page 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE If a security event affects immediate offsite access, notify 9-1-1 dispatcher to stage offsite assistance at Tri-County Marina.	
☐ 3.14 **IAAT** outside fire assistance is required, as determined by the Fire Brigade Leader / Incident Commander, **then** perform the following: ___ A. **CALL** 9-911 ___ B. **PROVIDE** address: TMI 1 Liberty Lane Londonderry Township ___ C. **NOTIFY** Security that offsite assistance has been requested.	
___ 3.15 **If** fire is in Unit 1 containment building, **then** perform the following: ___ A. **ACTUATE** RB Evacuation alarm. ___ B. **If** containment fire service spool piece is installed, **then OPEN** FS-V-367 (TB 305' west of CO-P-2s, 13' above floor) ___ C. If containment fire service spool piece is **<u>not</u>** installed, **then INITIATE** OP-TM-811-901, "Containment Fire Service," to place containment fire service system in service.	
___ 3.16 **REQUEST** SM to evaluate EALs and Events of Potential Public Interest.	
___ 3.17 **If** runoff from the event or fire fighting has the potential to reach the yard drain system, **then CLOSE** SD-V-120 (South of Unit 2 at East Dike).	
___ 3.18 **NOTIFY** the Station Fire Marshall of fire situation.	
___ 3.19 **When** the fire is out, **then GO TO** Section 4.0, Return to Normal.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.10	
	Importance Rating		3.9

(K&A Statement) Knowledge of conditions and limitations in the facility license.

Proposed Question: SRO 94

Plant conditions:

- The reactor was at 100% power when a grid disturbance caused Core Thermal Power (C1708) to stabilize at 2587 MWt.

Which ONE of the choices completes the following statement regarding the required action?

Immediately reduce core thermal power to less than __________.

A. 2581 and hold until the One Hour Average Core Thermal Power (C3560) is < 2581 MWt
B. 2581 and hold until the Four Hour Average Core Thermal Power (C3553) is < 2581 MWt
C. 2568 and hold until the One Hour Average Core Thermal Power (C3560) is < 2568 MWt
D. 2568 and hold until the Four Hour Average Core Thermal Power (C3553) is < 2568 MWt

Proposed Answer: D. 2568 and hold until the Four Hour Average Core Thermal Power (C3553) is < 2568 MWt

Explanation (Optional):

A. Plausible because this is the limit at which an immediate action is required but the one hour average applies to 2568.
B. Plausible because this is the limit at which an immediate action is required and the 4 hour average is correct.
C. Plausible because it is the correct limit but the one hour average is a different limit, not a corrective action.
D. Correct – immediate action to reduce to <2568 MWt and hold until 4 hour average is <2568 is required.

Technical Reference(s): 1102-4, 2.1.6 (Page 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	b.1	

Comments:

4. **If** the 4 hour average Core Thermal Power (C3553) exceeded 2568 MWt, **then EVALUATE** for an overpower condition IAW OP-AP-108-104-1001, PWR Overpower Assessment.

2.1.6 **If** core thermal power (C1708) ≥ 2581 MWt (100.5%), **then**:

1) **Immediately REDUCE** core thermal power to less than 2568 MWt (100%).

2) **MAINTAIN** reduced reactor power until four (4) hour best estimate core thermal power (C3553) is less than 2568 MWt.

2.1.7 **If** the heat balance calculation of core thermal power is not available (except for planned out of service of less than 1 hour), **then**:

1) **MAINTAIN** the highest operable power range NI (NI-5, 6, 7 and 8) less than 100%.

2) **If** last (i.e. prior to loss of CMS (Core Monitoring System) 4 hour best estimate core thermal power (C3553) was greater than 2568 MWt or can not be determined, **then**

i) **REDUCE** reactor power below 100% by 2% for every 1% that power was over 100%.

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%.

2.1.8 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.

2.1.9 Ensure that rate of reactor power increase is less than limit specified on Enclosure 1 (Mechanical Maneuvering Recommendations).

2.1.10 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.32	
	Importance Rating		3.8

(K&A Statement) Ability to explain and apply all system limits and precautions.

Proposed Question: SRO 95

Plant conditions:

- An automatic reactor trip occurred after 200 days of operation at full power.
- The crew is performing a reactor startup in accordance with 1103-8, "APPROACH TO CRITICALITY".
- ECP Data:
  - Estimated critical rod position is 90% withdrawn on CRG 6.
  - Minimum withdrawal limit is 40% withdrawn on CRG 6.
  - Maximum withdrawal limit is 50% withdrawn on CRG 7.

Current conditions:

- 50% withdrawn on CRG 7
- CRO reports and STA confirms that the 1/M predicts criticality at 65% withdrawn on CRG 7.

As CRS, which ONE of the following actions must be directed?

A. Commence emergency boration and contact Reactor Engineering.
B. Insert regulating rods to achieve at least a 1% ΔK/K subcritical condition.
C. Insert the rods to 90% withdrawn on CRG 6 and deborate to achieve criticality.
D. Leave the rods at 50% withdrawn on CRG 7, continue monitoring the NI's and re-evaluate the ECP.

Proposed Answer: B. Insert rods to achieve at least a 1% ΔK/K subcritical condition.

Explanation (Optional):

A. Plausible because emergency boration would be correct for unanticipated criticality and contacting Reactor Engineering is correct.
B. Correct – required by 2.1.7, 1103-8.
C. Plausible because this would be acceptable if it was a planned evolution.
D. Plausible because the same limitation discusses an assessment.

Technical Reference(s): 1103-8, 2.1.7, Page 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # X QR5107-08-Q03
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 b.5

Comments:

2.1.7 The estimated critical position (ECP) calculated in accordance with 1103-15B (Reference 1.5) specifies the rod position tolerance band. If criticality occurs outside the specified band:

❶ An assessment **shall** be performed.

❷ Regulating rods **shall** be inserted to maintain at least a 1% Δk/k subcritical condition during the assessment.

❸ Entry into this procedure (1103-8) is a "planned evolution". Entry into 1203-10, Unanticipated Criticality (Ref. 1.10) is not required unless criticality outside the ECP tolerance band is uncontrollable.

2.1.8 The Nuclear Instrumentation **shall** be continuously monitored during any reactivity addition.

2.1.9 Reactor power **shall not** exceed 1% of rated power unless normal conditions are established for:

❶ Operating temperature

❷ Operating pressure

❸ Control rod configurations

2.1.10 Safety rod groups **shall** be fully withdrawn prior to entering approach to criticality (deboration or regulating rod withdrawal with the intent to reduce to less than 1% ΔK/K subcritical) with the following exceptions (T.S. 3.1.3.5):

1. **Inoperable rod** – the approach to criticality may continue if one of the safety rods is declared inoperable and Item 4 below is verified.

2. **Physics testing** – some of the safety rod groups may be inserted during physics testing, however, Item 4 below must be maintained.

3. **Exercising control rods** – rod exercise surveillance can be performed without the safety rod groups being fully withdrawn, however, Item 4 below must be maintained.

4. **Shutdown margin may not be reduced below 1% ΔK/K.**

2.1.11 This procedure shall not be used for the initial criticality following refueling (when 1550-01 and 1550-02 apply).

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		2
K/A #	G2.2.28	
Importance Rating		3.5

(K&A Statement) Knowledge of new and spent fuel movement procedures.

Proposed Question: SRO 96

Plant conditions:

- A refueling outage is in progress.
- You are the SRO Fuel Handling Supervisor.
- An assembly has been withdrawn and the bridge is starting to move away from the core.
- RMS alarm RM-A-2, REACTOR BUILDING, has actuated in the control room.
- The Shift Manager has informed you that it appears RM-A-2 is failed but, as a precautionary measure, all personnel will be directed to exit the Reactor Building.

In accordance with 1505-1, "REFUELING PROCEDURE", which ONE of the choices completes the following statement regarding required actions prior to exiting the Reactor Building?

Return to the core and insert the assembly into ______________________.

A. any available core location then move the bridge away from the core
B. any available core location; leaving the grapple engaged
C. an approved location then move the bridge away from the core
D. an approved location; leaving the grapple engaged

Proposed Answer: D. an approved location; leaving the grapple engaged

Explanation (Optional):

A. Plausible in that it contains one-half of each incorrect choice.
B. Plausible because the second half is correct.
C. Plausible in that the first half is correct.
D. Correct – procedure requirement is to place it in an approved location and to leave the grapple engaged.

Technical Reference(s): 1505-1, REFUELING PROCEDURE, Step 5.3.9/5.3.10 (Page 10) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 b.6

Comments:

5.3.4 Fuel assembly shall **NOT** be disengaged from grapple until proper seating has been verified by ZZ tape reading.

5.3.5 When inserting into core, the fuel assembly shall not be disengaged from grapple until count rate stability has been verified.

> **NOTE**
>
> Plant Computer trend recorder or local indicators may be used as alternate means of monitoring reactivity additions.

- **IF** count rate exceeds twice the base line count (as determined by strip chart, or other means), Core Load Engineer and SRO (Fuel Handling Supervisor) shall evaluate.

5.3.6 Whenever core geometry is being changed, core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available. (T.S. 3.8.2)

5.3.7 When core geometry is **NOT** being changed, at least one neutron flux monitor must be in service (T.S. 3.8.2).

5.3.8 **IF** fuel loading is interrupted, it may be necessary to redefine initial count rate prior to resuming fuel loading.

5.3.9 Fuel assemblies must be loaded into their final core locations.

- **IF** fuel assembly is to be placed in any other core location due to fuel handling problems, etc., fuel assembly shall be placed **ONLY** into core locations which are designated by approved core reload design to contain an assembly with equal or higher reactivity.
- Core Load Coordinator must approve any deviation from this reactivity requirement.

5.3.10 **IF** containment evacuation is required during fuel handling, regardless of cause, Fuel Handling Supervisor shall direct the following activities to be completed if time and conditions permit:

> **NOTE**
>
> **IF** evacuation caused by fuel damage, **DO NOT** handle damaged assembly any further unless needed for immediate safety concerns.

- Any irradiated fuel or control component being handled shall be inserted into designated storage location. DO NOT DISENGAGE GRAPPLE.
- Both upenders shall be lowered for transfer to Fuel Handling Building and place carriage control switches shall be placed to Fuel Pool position.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.33	
	Importance Rating		2.9

(K&A Statement) Knowledge of control rod programming.

Proposed Question: SRO 97

Plant conditions:

- The unit is in cold shutdown.
- CRD maintenance required patch point reconnection.
- As a result, OP-TM-622-221, "CONTROL ROD PROGRAM SPECIAL CHECK", must be performed.

Which ONE of the following identifies the position that must approve unlocking the Patch Panel(s)?

A. Control Room Supervisor.
B. Shift Manager.
C. Operations Director.
D. Plant Manager.

Proposed Answer: D. Plant Manager.

Explanation (Optional):

A. Plausible since the position has control over issuing keys.
B. Plausible since the position approves the performance of the procedure.
C. Plausible since the position must approve some actions not delegated to the Shift Manager.
D. Correct – per OP-TM-622-221, the Plant Manager must approve unlocking that cabinet.

Technical Reference(s): TS 3.5.2.6 (Attach if not previously provided)
OP-TM-622-221, 4.1.3 (Page 2)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	b.2	

Comments:

3.5.2.5 Control Rod Positions

a. Operating rod group overlap shall not exceed 25 percent ± 5 percent, between two sequential groups except for physics tests.

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

   1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 24 hours, and FQ(Z) and $F_{\Delta H}^{N}$ shall be verified within limits once every 2 hours, or power shall be reduced to $\leq$ power allowed by insertion limits.

   2. If regulating rods are .inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to $\geq 1\%$ $\Delta K/K$, and restore regulating rods to within restricted region within 2 hours or reduce power to $\leq$ power allowed by rod insertion limits.

c. Safety rod limits are given in 3.1.3.5.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Plant Manager.

3.5.2.7 Axial Power Imbalance:

a. Except for physics tests the axial power imbalance, as determined using the full incore system (FIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

   The FIS is operable for monitoring axial power imbalance provided the number of valid self powered neutron detector (SPND) signals in any one quadrant is not less than the limit in the CORE OPERATING LIMITS REPORT.

b. When the full incore detector system is not OPERABLE and except for physics tests axial power imbalance, as determined using the power range channels (out of core detector system)(OCD), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

c. When neither detector system above is OPERABLE and, except for physics tests axial power imbalance, as determined using the minimum incore system (MIS), shall not exceed the envelope defined in the CORE OPERATING LIMITS REPORT.

d. Except for physics tests if axial power imbalance exceeds the envelope, corrective measures (reduction of imbalance by APSR movements and/or reduction in reactor power) shall be taken to maintain operation within the envelope. Verify FQ(Z) and $F_{\Delta H}^{N}$ are within limits of the COLR once per 2 hours when not within imbalance limits.

3.3 Prerequisites:

NOTE: 1302-5.13, Control Rod Absolute and Relative Position Indication Equipment Surveillance Calibration, requires this surveillance (OP-TM-622-221) to be performed after 1302-5.13 is performed.

3.3.1 **VERIFY** 1302-5.13, Control Rod Absolute and Relative Position Indication Equipment Surveillance Calibration completed if scheduled. ____

3.3.2 **VERIFY** CRD (622) System in Plant Shutdown Mode IAW OP-TM-622-000, Control Rod Drive System ____

4.0 MAIN BODY

4.1 Prepare for cable checks as follows:

4.1.1 **ENSURE** Maintenance Supervisor performs the following:

- **MARK** 'N/A' any connection point on Attachments 7.3 through 7.7 that was **not** disturbed and does **not** need to be checked.
- **If** an entire Attachment is **not** required, **then MARK** its associated procedure performance steps as 'N/A'.

______________________________ Date: ________
Maintenance Supervisor signature

______________________________ Date: ________
SM concurrence signature

4.1.2 **If** Power patch verification (Attachment 7.3) is required, **then ENSURE** CRD system de-energized and properly tagged out of service. ____

4.1.3 **If** Power, Group, or PI Patch verification (Attachment 7.3, 7.4, or 7.5) is required, **then VERIFY** Plant Manger permission obtained to unlock patch panel(s). ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.2	
	Importance Rating		2.9

(K&A Statement) Knowledge of facility ALARA program.

Proposed Question: SRO 98

Conditions:

- Work will be performed in an area that has a general radiation dose of 90 mRem/hr.
- One worker can perform the job in 1 hour and 15 minutes.
- Two workers can perform the job in 45 minutes.
- RadPro estimates that it will take two technicians 20 minutes each to install temporary shielding that will reduce the dose rate to 30 mRem/hr.
- An engineering evaluation has been completed to allow shielding to remain if installed.

Which one of the following methods of performing the job meets the facility criteria for RP-AA-400, "ALARA PROGRAM"?

A. One worker with temporary shielding.
B. One worker WITHOUT temporary shielding.
C. Two workers with temporary shielding.
D. Two workers WITHOUT temporary shielding.

Proposed Answer: A. One worker with temporary shielding.

Explanation (Optional):

A. Correct – yields lowest total dose (RadPro 60 mr + Worker 37.5 mr = 97.5 mr).
B. Plausible with incorrect calculation (RadPro 0 + Workers 112.5 mr = 112.5 mr)
C. Plausible with incorrect calculation (RadPro 60 mr + Worker 45 mr = 105 mr).
D. Plausible with incorrect calculation (RadPro 0 + Workers 135 mr = 135 mr).

Technical Reference(s): RP-AA-400, 2.2 (Pg. 1) (Attach if not previously provided)
RP-AA-400, 3.10 (Pg. 5)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank X #	IS-OP-AA-RPT-Q01	(Note changes or attach parent) Changed values, format, order of choices.
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	b.4	

Comments:

ALARA PROGRAM

1. PURPOSE

1.1. This procedure establishes the requirements and responsibilities for the effective implementation of the ALARA Program. The objective of the ALARA program is to ensure that occupational radiation exposure, both individually and collectively, is maintained ALARA.

2. TERMS AND DEFINITIONS

2.1. **ALARA**: Acronym for "as low as reasonably achievable." ALARA means making every reasonable effort to maintain exposure to radiation as far below the dose limits, as defined in 10 CFR 20, as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed material in the public interest.

2.2. **ALARA Plan**: A documented job evaluation. The evaluation will consider the radiological conditions expected during each phase of the job and the methods and controls to minimize contamination and collective radiation exposure (person-rem). The term "ALARA Plan" is equivalent to "ALARA Action Review" and "ALARA Review."

2.3. **Station ALARA Committee (SAC)**: Station committee responsible for the overall coordination of the Station ALARA Program and for advising site management in matters relating to ALARA and other pertinent Radiation Protection programs.

3. RESPONSIBILITIES

NOTE: Sites may utilize position titles that are **not** identical to those outlined in this procedure. In those cases, the responsibilities should be performed by the site individual with responsibilities closest to those described in this procedure.

3.1. Plant Manager/Site Vice President

3.1.1. **ACT** as the Chairperson for the Station ALARA Committee under normal circumstances. If the Plant Manager or Site Vice President is **not** available to fulfill this duty, a senior member of station management staff may be appointed as an alternate to chair the meeting.

3.9.13. **SERVE** as the SAC Secretary **or DESIGNATE** an individual to serve as SAC Secretary.

3.10. "Task Manager" (Work Supervisor/ Project Manager)

3.10.1. **ENSURE** that department personnel comply with ALARA Program procedures and requirements.

3.10.2. **PROVIDE** input, including worker level involvement whenever possible, to the ALARA planning process.

3.10.3. **PRESENT** ALARA Plans, with the assistance of Radiological Engineering, with exposure estimates greater than or equal to 5 person-rem and those selected by SAC or identified by the Rad Engineering Manager as requiring a higher level of review based on the nature of the work or industry data.

3.10.4. **INTEGRATE** the ALARA concept into appropriate department procedures, work plans, and practices.

3.10.5. **REVIEW** exposure status of personnel under their supervision.

3.10.6. **MONITOR** exposure accumulation of jobs/ tasks being supervised.

3.10.7. **PROVIDE** input for Work-In-Progress (WIP) and ALARA Post-Job Reviews when required.

3.10.8. **MONITOR** in field performance as it relates to the execution of ALARA Plans and radiation protection practices **and PROVIDE** feedback to Radiological Engineering.

4. MAIN BODY

4.1. Station ALARA Committee Charter

4.1.1. A Station ALARA Committee (SAC) is established with responsibility for overall coordination of the Station ALARA Program and for advising site managers/directors in matters relating to ALARA.

4.1.2. The SAC membership is comprised of the following voting positions:

1. Plant Manager and/or Site Vice President
2. Radiation Protection Manager
3. Engineering Director
4. Operations Director
5. Maintenance Director
6. Chemistry Director

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G2.3.1	
	Importance Rating		3.0

(K&A Statement) Knowledge of 10 CFR: 20 and related facility radiation control requirements

Proposed Question: SRO 99

Plant conditions:

- The unit is at 100% power.
- Two NLO's must enter a Level 2 Locked High Radiation Area to verify the position of a manual valve that is not documented in the proper position.
- The job is expected to take only 1-2 minutes and total exposure for each person is expected to be < 100 mrem.

Which one of the following positions must provide approval prior to the entry?

A. Shift Manager.
B. Duty RadCon Technician.
C. Radiation Protection Manager.
D. Plant Manager or Site Vice President.

Proposed Answer: C. Radiation Protection Manager.

Explanation (Optional):

A. Plausible because the Shift Manager would be involved for an emergency exposure.
B. Plausible because it requires a specific RWP.
C. Correct – per RP-AA-460, prior approval is required from the Radiation Protection Manager.
D. Plausible because the Plant Manager or Site VP must approve VHRA entry.

Technical Reference(s): RP-AA-460, 4.4.2 (Page 6) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ______

	Modified Bank #	______	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	______	
10 CFR Part 55 Content:	55.41	______	
	55.43	b.4	

Comments:

Oyster Creek
The ability to control access to an LHRA through the use of a barricade and red flashing light in step 4.3.3 is **not** applicable at Oyster Creek.

4.3.3. **If no** enclosure exists for purposes of locking an LHRA located within a large area such as containment, and an enclosure **cannot** reasonably be constructed, **then**:

1. **OBTAIN** the written approval of the Radiation Protection Manager, or designee, using Attachment 4 for use of a barricade and red flashing light to control access, **and**

2. **BARRICADE and** conspicuously **POST** the area in accordance with RP-AA-376, **and ACTIVATE** a red flashing light(s) as a warning device.

3. **USE** a ladder lock, if appropriate, to control access to an LHRA. **POST** the ladder as an LHRA in accordance with RP-AA-376.

4.4. Controls Level 2 LHRA (> 15,000 mrem/hr at 30 cm)

4.4.1. **FOLLOW** the requirements in Section 4.3.

4.4.2. **If** entry into dose equivalent rates >15,000 mrem/hr at 30 cm is required (and for other areas designated by the Radiation Protection Manager as Level 2 LHRAs), **then OBTAIN** approval of the Radiation Protection Manager, or designee, prior to entry. The approval may be obtained by telephone.

1. **DOCUMENT** approval on Attachment 5, Access to Level 2 LHRA or VHRA Approval Form.

4.5. Requirements for entry into a HRA or LHRA

4.5.1. **AUTHORIZE** access to an HRA or LHRA through the issuance of a Radiation Work Permit (RWP) that includes appropriate provisions for radiation protection equipment and exposure controls measures.

1. All RWPs that allow access to HRA or LHRA areas shall include the requirement of each individual to be issued an electronic dosimeter.

4.5.2. To gain access to an HRA or LHRA, **PERFORM** actions in accordance with the responsibilities outlined in Section 4.8.

4.5.3. **REQUIRE** individuals accessing an HRA or LHRA to be equipped with one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G2.4.41	
	Importance Rating		4.1

(K&A Statement) Knowledge of the emergency action level thresholds and classifications.

Proposed Question: SRO 100

Plant conditions:

- The operating crew has just initiated a plant shutdown due to OTSG "A" tube leakage of ~~1.5 gpm.~~ 125 gpm to the Condenser Sret
- The Shift Manager has validated a report from NRC that a Boeing 737 was hijacked in Chicago. The FAA informed the NRC that the hijacker has directed the pilot to fly to Harrisburg International Airport for refueling and further instructions.
- The NRC notified TMI that it considers this to be a direct threat to TMI until further notice.
- The plane is in the air and 22 minutes from Harrisburg.

Which one of the following is the correct Emergency Action Level (EAL) for this event?

A. UNUSUAL EVENT.
B. ALERT.
C. SITE AREA EMERGENCY.
D. GENERAL EMERGENCY.

Proposed Answer: B. ALERT.

Explanation (Optional):

A. Plausible because it involves aircraft threats but not within 30 minutes.
B. Correct – large plane, verified threat, <30 minutes away. OTSG tube leakage is < EAL threshold.
C. Plausible because it involves large aircraft impact.
D. Plausible because of the potential for the loss of physical security.

Technical Reference(s): EAL MATRIX, Section H (Attach if not previously provided)
EAL Basis, HA-1

Proposed references to be provided to applicants during examination: EAL MATRIX

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	b.5	

Comments:

RECOGNITION CATEGORY
HAZARDS AND OTHER CONDITIONS

HA1

INITIATING CONDITION

Notification of an Airborne Attack Threat

EAL THRESHOLD VALUES

1. A validated notification from NRC of a LARGE AIRCRAFT attack threat < 30 minutes away.

MODE APPLICABILITY

ALL Plant Conditions

BASIS (References)

LARGE AIRCRAFT: Aircraft as large as or larger than passenger airliners or air cargo / freight planes (for example; 737, DC9, MD80, MD90, 717 or C-130). Examples of aircraft that would not be considered large are general aviation Cessna, Piper and Lear type private planes as well as police, medical and media helicopters.

The intent of this EAL is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This EAL is met when a plant receives information regarding a LARGE AIRCRAFT attack threat from NRC and the LARGE AIRCRAFT is less than 30 minutes away from the plant.

This EAL is intended to address the contingency for a very rapid progression of events due to an airborne hostile attack such as that experienced on September 11, 2001. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from such an attack. Although vulnerability analyses show Nuclear Power Plants to be robust, it is appropriate for Offsite Response Organizations to be notified and encouraged to activate (if they do not normally) to be better prepared should it be necessary to consider further actions. LARGE AIRCRAFT is meant to be an aircraft with the potential for causing significant damage to the plant. The status and size of the plane may be provided by NORAD through the NRC.

TABLE TMI 3-1: Emergency Action Level (EAL) Matrix (Cont'd)

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT	
HAZARDS AND OTHER CONDITIONS					
Terrorism Security Events	**HG1 (Applicability: ALL Plant Conditions)** *Security Event Resulting in Loss of Physical Control of the Facility.* EAL Threshold Value: A HOSTILE FORCE has taken control of the following: 1. Plant equipment such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. Control of spent fuel pool cooling systems if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in pool).	**HS1 (Applicability: ALL Plant Conditions)** *Site Attack* EAL Threshold Value: 1. A notification from the site security force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA	**HA1 (Applicability: ALL Plant Conditions)** *Notification of an Airborne Attack* EAL Threshold Value: 1. A validated notification from NRC of a LARGE AIRCRAFT attack threat < 30 minutes away.	**HU1 (Applicability: ALL Plant Conditions)** *Confirmed Terrorism Security Event That Indicates a Potential Degradation in Level of Plant Safety* EAL Threshold Value: 1. A credible site specific security threat notification as determined per SY-AA-101-132, "Threat Assessment." **OR** 2. A validated notification from NRC providing information of an aircraft threat.	Terrorism Security Events
Terrorism Security Events	None	None	**HA2 (Applicability: ALL Plant Conditions)** *Notification of HOSTILE ACTION within the OCA.* EAL Threshold Value: 1. A notification from the site security force that an armed attack, explosive attack, LARGE AIRCRAFT impact, or other HOSTILE ACTION is occurring or has occurred within the OCA.	None	Terrorism Security Events
Non Terrorism Security Events	None	**HS3 (Applicability: ALL Plant Conditions)** *Security Event in a Plant Vital Area.* EAL Threshold Value: 1. Security event as determined from the Station Security Plan – Appendix C in a plant VITAL AREA and reported by the Security Force	**HA3 (Applicability: ALL Plant Conditions)** *Security Event in a Plant Protected Area* EAL Threshold Value: 1. Security event as determined from the Station Security Plan – Appendix C in a plant PROTECTED AREA and reported by the Security Force.	**HU3 (Applicability: ALL Plant Conditions)** *Confirmed Security Event That Indicates a Potential Degradation in Level of Plant Safety* EAL Threshold Value: 1. Security event as determined from the Station Security Plan – Appendix C and reported by the Security Force.	Non Terrorism Security Events
Control Room Evacuation	None	**HS4 (Applicability: ALL Plant Conditions)** *Control Room Evacuation Initiated AND Plant Control CANNOT be re-established in ≤ 15 minutes* EAL Threshold Value: 1. Evacuation of the Control Room has been **INITIATED** **AND** **ALL** of the following have **NOT** been performed **≤ 15 minutes** of the evacuation: • Protected supply of electrical power established or available **AND** • Protected supply of RCS make-up, letdown and seal injection is established **AND** • Primary-to-Secondary heat transfer is established and controlled	**HA4 (Applicability: ALL Plant Conditions)** *Control Room Evacuation is Initiated* EAL Threshold Value: 1. Evacuation of the Control Room is **INITIATED**	None	Control Room Evacuation