Question #: **1** | **Question ID: 9000018** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 1 | Rev. 0 | ☑ Selected for Exam | **Origin: New** | ☐ Past NRC Exam?

The plant automatically tripped on High Pressurizer Pressure due to an inadvertent closure of the Main Turbine Control Valves.

During the performance of EOP 2525, Standard Post Trip Actions, the crew reported that Bus 24D is deenergized due to a fault <u>and</u> that Power Operated Relief Valve (PORV), RC-404, is stuck open. All other equipment operated as designed. Upon entry into EOP 2532, Loss of Coolant Accident, the following conditions exist:

- Containment pressure is 4.5 psia and slowly rising.
- Reactor vessel is 43% and slowly going down
- CET temperatures are 578°F and stable
- RCS pressure is 1310 psia and stable
- Pressurizer level is 100%.
- Steam generator levels are both 41% and going up slowly.

Which of the following actions must the Unit Supervisor/Shift Manager perform to preserve a Safety Function?

☐ **A** Direct the Technical Support Center to develop a plan to restore RWST level.

☐ **B** Direct the Balance of Plant Operator to align Condenser Air Removal to the Unit 2 Stack.

☐ **C** Direct the Reactor Operator to place the SI/CS Pump Miniflow switches in "OPERATE".

☑ **D** Direct the crew to commence a controlled cooldown and depressurization.

Justification

D IS CORRECT; With RCS pressure stable at 1310 psia and the PORV still open, RCS inventory is being lost faster than Charging can restore it. The steps for the cooldown and subsequent depressurization must be pulled forward (performed out of sequence) to allow RCS pressure to be reduced below HSPI shut off head to allow adequate Safety Injection flow.

A is incorrect; Although RWST level is lowering, there is NO need to develop a plan to restore RWST level at this time (perform step out of sequence).
<u>Plausible</u> because step 8 of EOP 2532 directs the US or SM to have the TSC develop a plan for restoring level in the RWST if the LOCA is determined to be outside of Containment. Examinee may not remember that this step is performed ONLY if the LOCA is outside of Containment.

B is incorrect; With Containment pressure >4.42 psia, MSI has actuated and the MSIVs are closed resulting in a loss of Condenser vacuum; therefore, the is no need to align Condenser Air Removal to the Unit 2 stack.
<u>Plausible</u> because this is a procedurally directed step. Step 15 states, "If EBFAS has initiated and the Condenser is available, then align Condenser Air Removal to Unit 2 stack." If the examinee does not realize that MSI has actuated, then this step may be performed out of sequence.

C is incorrect; The SI/CS Pump Miniflow switches are not placed in 'OPERATE" until RWST level is ≤20%.
<u>Plausible</u> because the examinee may feel that a Sump Recirc Actuation Signal is imminent; therefore, it would be appropriate to perform this step out of sequence.

References

EOP 2532, "LOCA" and OP 2260, "EOP Users Guide"

Comments and Question Modification History

Bob K. - D-3/C, K
Corrected typo in distractor B.
Bill M. - D-2/C, K
Angelo - D-2/C; No comments.

NRC K/A System/E/A **System** 009 Small Break LOCA

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.20 **RO** 4.6 **SRO** 4.6 **CFR Link** (CFR: 41.10 / 43.5 / 45.12)

Ability to interpret and execute procedure steps.

Question #: 1 | Question ID: 9000018 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
I-SRO Ques. # 1 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Millstone Unit 2 Loss of Coolant Accident	EOP 2532 Revision 029 Page 21 of 95

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

1. RCS cooldown should be initiated within one hour after the event to conserve condensate inventory and comply with the Long Term Cooling Analysis.
2. RCS cooldown rate greater than 40°F/hr should be maintained until the steam dump/bypass valves or atmospheric dump valves are full open.
3. The starting point for the RCS cooldown should be the T_C or CET temperatures where RCS has stabilized.
4. T_C should be used for monitoring RCS cooldown if in forced or natural circulation. CETs should be used for all other cases.

NOTE

Technical Specification cooldown rates should be observed during the cooldown. The cooldown rates are as follows:

1. RCS T_C greater than 220°F the cooldown rate is 100°F/hr.
2. RCS T_C less than or equal to 220°F the cooldown rate is 50°F/hr.

Perform Controlled Cooldown

INSTRUCTIONS	CONTINGENCY ACTIONS
*17. INITIATE a controlled cooldown using the steam dumps to establish shutdown cooling entry conditions.	17.1 INITIATE a controlled cooldown using the ADVs to establish shutdown cooling entry conditions.

STOP THINK ACT REVIEW

b. Procedural steps listed in alphanumeric order are sequential steps and shall be addressed in that sequence. Exceptions to this are as follows.

- Asterisked steps, within the ORP or selected FRPs being implemented, may be brought forward to correct or preserve a Safety Function.
- Steps may be performed out of order after they have been accomplished once, if they are Continuously Applicable step, as indicated by an asterisk.

c. Steps with recurrent actions (i.e., the step will be performed repeatedly during the procedure) should be checked off in the placekeeper when started. Since these steps will be performed repeatedly, the placekeeper is marked with a "cont," in the "Done" column. This signifies that the step is continuously performed and will not be completed during the performance of the EOP.

d. Bulleted lists are provided within a step when any one of several alternative actions are equally acceptable to perform. The preferred method is listed as the first alternative.

e. It is acceptable for the SM or US to direct the performance of a task out of sequence, if the actions do not interfere with maintaining an existing Safety Function (e.g., transferring power supplies for "B" charging pump, preparing for power restoration).

1.9.2 Instructions and Contingency Actions

a. The EOPs are formatted with Instructions and Contingency Actions. The instructions column presents the optimal method and sequence for accomplishing a specific task. The contingencies column contains actions to be performed if the optimum method cannot be accomplished.

b. If the expected response is obtained (left column), the operator proceeds to the next step or sub step in the Instructions column (left column).

Question #:	2	Question ID:	9000019	☐ RO	☑ SRO	☐ Student Handout?	☐ Lower Order?
I-SRO Ques. #	2	Rev.	0	☑ Selected for Exam	Origin:	New	☐ Past NRC Exam?

The plant tripped from 100% power due to a Large Break LOCA. The crew successfully completed all actions of EOP 2525, Standard Post Trip Actions, and are presently performing EOP 2532, Loss of Coolant Accident.

The following conditions exist approximately 2 hours after the trip:
* SRAS actuated approximately 15 minutes ago.
* Containment pressure is 5 psig and slowly lowering.
* RCS pressure is 360 psia and slowly lowering.
* CET temperatures indicate 434°F and slowly lowering.
* HPSI Pump current and flow are fluctuating.

Which of the following describes the cause of the HPSI Pump current and flow fluctuations, and the initial action that must be directed?

☐ **A** Hot water in the Containment Sump is flashing to steam in the HPSI Pump suctions.
Start at least 2 CAR Fans in Fast speed.

☑ **B** The HPSI Pumps are showing signs of cavitation due to Containment Sump clogging.
Secure both Containment Spray Pumps.

☐ **C** Boron is beginning to plate out in the core causing alternately high and low HPSI flow.
Establish Hot Leg and Cold Leg Injection.

☐ **D** Total Safety Injection flow is higher than necessary for the present conditions.
Throttle the HPSI Injection valves as needed.

Justification

B IS CORRECT; Sump clogging will cause a lower suction pressure in all the running SI pumps. A lower suction pressure will cause the HPSI Pumps to cavitate. EOP 2532 directs the CS pumps be secured (if not needed) to limit the competition for sump suction flow. EOP 2532 also requires Containment Spray Pumps to remain in operation for at least 4 hours for Iodine scrubbing; however, core cooling (maintaining adequate SI flow to the core) takes precedence over Iodine scrubbing.

A is incorrect; The Containment Spray System and CAR Coolers are designed to lower containment sump temperature enough to prevent cavitation of the HPSI Pumps.
Plausible because the SRAS caused the HPSI Pump suctions to swap from the RWST (cool water) to the Containment Sump (hot water). Starting 2 CAR Fans in Fast speed would help to lower the Containment Sump temperature; however, there is no procedural guidance to perform this action.

C is incorrect; Boron Precipitation is analyzed to occur after a large break LOCA; however, it is NOT analyzed to occur until 8-10 post-event. It's NOT likely that Boron would be solidifying in the core at this time.
Plausible because accident analysis shows boron Precipitation will occur after a large break LOCA. The examinee may NOT remember the time frame for Boron Precipitation (8-10 hours after the LOCA). Simultaneous Hot Leg and Cold Leg Injection is the appropriate action for Boron Precipitation.

D is incorrect; Total Safety Injection flow is likely above the SI flow curve. The curve is based on having only one train of SI in service (Accident Analysis). However, the additional flow does NOT adversely impact core cooling.
Plausible because the examinees should know that Safety Injection flow is higher than required for core cooling and may conclude that throttling HPSI is appropriate. EOP 2532 has steps for throttling HPSI flow.

References

EOP-2532, St. 50, Indications of CTMT Sump Clogging

Comments and Question Modification History

Bob K. - D-4/W (talked self out of answer)
Corrected typo in distractor D.
Bill M. - D-2/C, K
Angelo - D-4/C. Fair question.

NRC K/A System/E/A System 011 Large Break LOCA

Number EA2.10 **RO** 4.5 **SRO** 4.7 **CFR Link** (CFR 43.5 / 45.13)

Ability to determine or interpret the following as they apply to a Large Break LOCA: Verification of adequate core cooling

Question #: 2 | Question ID: 9000019 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
I-SRO Ques. # 2 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Millstone Unit 2 Loss of Coolant Accident	EOP 2532 Revision 029 Page 42 of 95

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Degradation in HPSI pump performance, post SRAS, may be indicative of debris fouling the CTMT sump screen. Checking HPSI pump flow greater than 30 gpm ensures minimum flow requirements are met for pump protection when RCS pressure is high and prohibiting flow. This presents differently than the sump blockage issue.

HPSI Pump Post SRAS Performance Criteria

*50. IF SRAS has actuated, CHECK for adequate HPSI flow by observing ALL of the following:

- Flow greater than or equal to 30 gpm for each operating pump
- Motor current stable
- Stable HPSI pump discharge pressure

50.1 IF unable to maintain HPSI flow due to high RCS pressure, STOP ONE HPSI pump to establish the following for the operating HPSI pump:

- Flow greater than or equal to 30 gpm
- Motor current stable
- Stable HPSI pump discharge pressure

50.2 IF HPSI pump performance degradation is due to CTMT sump clogging, (suction problem) PERFORM the following, as necessary, to attempt restoration of HPSI flow:

a. IF CTMT pressure can be maintained less than 54 psig, AND at least ONE complete facility of CAR fans is operating, STOP CS pumps.

(continue) (continue)

STOP THINK ACT REVIEW

Question #: 3	***Question ID:*** ***9000003***	☐ **RO**	☑ **SRO**	☐ **Student Handout?**	☐ **Lower Order?**
I-SRO Ques. # 3	**Rev.** **0**	☑ **Selected for Exam**	***Origin:*** ***New***	☐ **Past NRC Exam?**	

While operating at 100% power, the RCP A UPPER SEAL PRES HI annunciator alarms. While referring to the appropriate Annunciator Response Procedure, the RCP A BLEED-OFF FLOW HI annunciator alarms. Within a minute, the RCP A BLEED-OFF FLOW HI annunciator clears and the RCP A BLEED-OFF FLOW LO annunciator alarms and remains lit. Numerous annunciators associated with "A" RCP seals also alarm.

Which of the following describes the reason for this sequence of annunciators <u>and</u> the direction that must be given?

☑ **A** The "A" RCP Excess Flow Check Valve has seated. Manually trip the reactor and turbine, then stop the "A" RCP.

☐ **B** The "A" RCP Middle Seal has failed. Evaluate the condition of the other seals to confirm no other degradation or failures.

☐ **C** The Bleedoff Pressure Controller, PIC-215, has malfunctioned. Using the Foxboro Controller, restore "A" RCP Bleedoff pressure and flow to the normal band.

☐ **D** The RCP Bleedoff Relief Valve has inadvertently opened. Evaluate "A" RCP Seal pressures to determine whether or not the "A" RCP may remain in operation.

Justification

A is CORRECT; The RCP A UPPER SEAL PRES HI annunciator is indicative of a failure of the "A" RCP Middle or Lower Seal (or a combination of both). This resulted the a high Bleedoff flow through the "A" RCP Seals resulting in a RCP A BLEED-OFF FLOW HI annunciator. At 10 gpm. the Excess Flow Check Valve will close causing the RCP A BLEED-OFF FLOW HI annunciator to clear and the RCP A BLEED-OFF FLOW LO to annunciate ((0.75 gpm). At this point, the RCP Seal package has NO cooling flow and the RCP must be tripped. Procedurally, the reactor and turbine are tripped prior to tripping the affected RCP.

B is incorrect; The individual indications provided could be a result of a failure of the "A" RCP Middle Seal; however, the indications together would indicate a loss of Bleedoff flow through the "A" RCP seals requiring the RCP to be stopped. If a Middle Seal had failed, then the action is correct. If Bleedoff Pressure Controller, PIC-215, had malfunctioned, then the action would be appropriate.
<u>Plausible</u> because the Annunciator Response Procedures for the Upper Seal High Pressure and Bleedoff High Flow annunciators state that these alarms may be indicative of a failed Middle Seal.

C is incorrect; The above indications could be indicative of a failure of the RCP Bleedoff Pressure Controller; however, **all 4** RCPs would have similar annunciators. Additionally, the RCP Bleedoff Relief Valve would cycle to provide flow through **all 4** of the RCP Seals. Use of the Foxboro Controller will NOT be effective. If the RCP Bleedoff Relief Valve has inadvertently opened, then the action would be appropriate.
<u>Plausible</u> because a failed or failing RCP Bleedoff Pressure Controller could give the above indications; however, all 4 RCPs would be affected.

D is incorrect; Opening of the RCP Bleedoff Relief Valve would likely result in a high Bleedoff flow on **all 4** RCPs and NOT a low Bleedoff flow annunciator. Additionally, "A" RCP Seal pressures are dependent on seal conditions and not necessarily the flow path of Bleedoff.
<u>Plausible</u> because the examinee may be confused on how the Bleedoff Relief Valve would impact an individual RCP Bleedoff flow and pressure. The examinee may also be confused as to whether each RCP had a bleedoff relief valve.

References

ARP 2590B-068

Comments and Question Modification History

Bob K. - D-3/C
Bill M. - D-3/C, 50/50
Angelo - D-4/C; No comments.

NRC K/A System/E/A **System** 015 Reactor Coolant Pump Malfunctions

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.7 **RO** 4.4 **SRO** 4.7 **CFR Link** (CFR: 41.5 / 43.5 / 45.12 / 45.13)

Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Question #: 3 | Question ID: 9000003 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 3 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Approval Date 5/26/09 Effective Date 6/4/09

Setpoint: 2 gpm

BB-17

RCP A BLEED-OFF FLOW HI

AUTOMATIC FUNCTIONS

1. None

NOTE

This alarm may be indicative of seal stage failure. One seal failure can cause high bleedoff flow alarm. Operation may continue with this alarm present providing seal bleedoff temperature is within limits and seal differential pressures indicates that only one of the three lower seal stages failed. A vapor stage failure will require a plant trip. ②

CORRECTIVE ACTIONS

NOTE

If seal flow reaches 10 gpm, "A" RCP controlled bleedoff excess flow check valve closes to prevent blockage of bleedoff flow from other RCPs.

5. Refer To the following guidance and DETERMINE if "A" RCP controlled bleedoff excess flow check valve has closed:
   - High bleedoff flow alarm annunciates and clears followed by low bleedoff flow alarm which remains lit
   - Seal pressures rise on *all* 3 seals (provided vapor seal is intact)
6. IF "A" RCP controlled bleedoff excess flow check valve has closed, PERFORM the following:
   - 6.1 TRIP reactor and turbine.
   - 6.2 STOP "A" RCP.
   - 6.3 Go To EOP 2525, "Standard Post Trip Actions" and PERFORM required actions.

Question #: **4** | **Question ID: 9000004** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

I-SRO Ques. # **4** | Rev. **0** | ☑ **Selected for Exam** | ***Origin:*** ***New*** | ☐ **Past NRC Exam?**

The plant is operating at 100% power with the "B" Auxiliary Feedwater (AFW) Pump out of service for maintenance.

Then the following events occur:

* Automatic plant trip due to a Steam Generator Tube Rupture (SGTR) on #2 Steam Generator (SG).
* Loss of the RSST and VA-20 at the time of trip.
* Shortly after the trip, a Safety Injection Actuation signal (SIAS) automatically actuated.
* All other plant systems respond as designed.

Which one of the following actions must the US perform during EOP 2525, Standard Post Trip Actions, to mitigate the consequences of this event and what is the reason for this action?

☐ **A** Dispatch a PEO to the Hot Shutdown Panel, C-21, to throttle open the #2 Atmospheric Dump Valve. This will permit a cooldown of both Hot Leg temperatures to ≤ 515°F.

☐ **B** Direct the BOP to swap the control power supply switch for the Terry Turbine to Facility 1. This will allow the operator to maintain both S/G levels in the prescribed bands.

☑ **C** Dispatch a PEO to manually operate the "B" Auxiliary Feedwater Regulating Valve, 2-FW-43B. This will prevent excessive auxiliary feedwater from overfilling the affected SG.

☐ **D** Direct the BOP to close #2 S/G Steam Supply to the Terry Turbine, MS-202, after the disconnect is closed. This will minimize the radioactive release from the affected SG.

Justification

C IS CORRECT; On a loss of normal power, Condensate is lost; therefore, main Feedwater is lost. The loss of VA-20 will cause the "B" Aux Feed Regulating Valve to fail open. EOP 2525 requires at least two Auxiliary Feed Pumps to be started. In order to prevent overfeeding #2 SG, the "B" Aux Feed Regulating Valve, 2-FW-43B, must be either closed or isolated locally.

A is incorrect; A loss of VA-20 will result in a loss of power to the #2 ADV from ALL remote locations. The #2 ADV can ONLY be operated locally with the handwheel in manual. (A loss of VR-21 will result in the loss of control to the #2 ADV from C-05. The ADV may then be controlled from C-21.)
Plausible because the examinee may think that the Facility 2 components controlled from Hot Shutdown Panel, C-21 are powered from VR-21 or VA-40 and are NOT affected by a loss of VA-20.

"B" is incorrect; Control power supply to the Turbine Driven Auxiliary Feedwater Pump is from DV-20, NOT VA-20; therefore, swapping power supplies will have NO impact on the availability of the Turbine driven Auxiliary Feedwater Pump.
Plausible because the examinee may not remember that the power supply for the TDAFP is DV-20 NOT VA-20

"D" is incorrect; #2 S/G Steam Supply to the Terry Turbine, MS-202, will be closed to minimize the release of radioactive steam from the Terry Turbine exhaust; however, this action CANNOT be performed in EOP 2525. This action is only performed in EOP 2534, after lowering both hot leg temperatures to <515°F, when isolating the affected S/G.
Plausible because this action will be performed at a later time and for the stated reason.

References

EOP 2525, AOP 2504D.

Comments and Question Modification History

Bob K. - D-4/C
Bill M. - D-2/C, K
Angelo - D-4/C; **Change "close" to "operate" in correct answer. - RLC**

NRC K/A System/E/A | **System** 038 Steam Generator Tube Rupture (SGTR)

Generic K/A Selected

NRC K/A Generic | **System** 2.1 Conduct of Operations

Number 2.1.30 **RO** 4.4 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.7)

Ability to locate and operate components, including local controls.

Millstone Unit 2 Loss of 120 VAC Vital Instrument Panel VA-20 [♣]	AOP 2504D Revision 003-07 Page 3 of 22

1.0 PURPOSE

1.1 Objective

This procedure provides instructions to be performed upon the loss of 120 Volt AC Vital Instrument Panel VA-20.

1.2 Discussion

The loss of VA-20 causes the loss of many annunciators, indications, PPC inputs, interlocks and controls. Equipment that affects Unit operation includes the following:

- FW-51B, #2 SG FRV, fails as is on loss of power, and closes when power is restored
- FW-41B, #2 SG FRV bypass, closes and control from C-05 is lost
- "C" charging pump loses the plunger flush pump
- If channel "Y" is controlling pressurizer level, charging will go to maximum and letdown will go to minimum
- If pressurizer pressure control is selected to channel "Y," auto control of pressurizer sprays are lost
- All pressurizer heaters are deenergized due to pressurizer level channel Y sending a pressurizer level low low heater cutout. To restore pressurizer heaters, the heater select switch must be placed in the channel "X" position.
- #2 SG "ATMOS DUMP MS 190B" control from C-05, C-21 and C-10 is lost.
- Steam dump to condenser pressure controller (PIC-4216) from C-05 is lost. Control from Foxboro computer is available.
- #2 Aux feed reg valve fails open and manual control from C-05 is lost.

Millstone Unit 2 Standard Post Trip Actions	EOP 2525 Revision 023 Page 16 of 26

INSTRUCTIONS	CONTINGENCY ACTIONS
___6. (continued)	
c. CHECK that at least one steam generator has **BOTH** of the following conditions met: • Level is 10 to 80%. • Main feedwater or TWO auxiliary feedwater pumps are operating to restore level 40 to 70%.	c.1 RESTORE level to 40 to 70% in at least one steam generator using ANY of the following: • Main feedwater • Motor–driven auxiliary feedwater pump • TDAFW Pump. Refer To Appendix 6, "TDAFW Pump Normal Startup." • TDAFW Pump. Refer To Appendix 7, "TDAFW Pump Abnormal Startup."
d. CHECK that RCS subcooling is greater than or equal to 30°F.	d.1 RESTORE steam generator level 40 to 70% by performing ONE of the following: • FEED each unaffected steam generator greater than 300 gpm. • FEED the least affected steam generator greater than 300 gpm.

Millstone Unit 2 Standard Post Trip Actions	EOP 2525 Revision 023 Page 18 of 26

INSTRUCTIONS	CONTINGENCY ACTIONS
__7. (continued)	
b. CHECK that **NONE** of the following steam plant radiation monitors have an unexplained alarm or indicate an unexplained rise in activity: **Steam Plant Radiation Monitors** • RM−5099, Steam Jet Air Ejector • RM−4262, SG Blowdown • RM−4299A and B, Main Steam Line 1 • RM−4299C, Main Steam Line 2	b.1 <u>IF</u> feed is available to **BOTH** steam generators, THROTTLE feed to the steam generator with the highest radiation readings to maintain level 40 to 45% by performing **ANY** of the following: 1) OPERATE associated main feed reg bypass valve, FW−41A or FW−41B. 2) <u>IF</u> AFAS has actuated, PERFORM the following: • PLACE the auxiliary feed "OVERRIDE/ MAN/START/ RESET" handswitches in "PULL TO LOCK". • OPERATE the associated aux feed reg valve, FW−43A or FW−43B.
(continue)	(continue)

Millstone Unit 2 Emergency Operating Procedure Technical Guide	EOP 2525, Standard Post Trip Actions Page 19 of 38

Step Number 7 Determine Status of Containment Isolation

The intent of the Containment Isolation safety function is to ensure that containment atmospheric conditions are acceptable or that mitigative actions are initiated.

INSTRUCTIONS

Containment Isolation serves to ensure that radioactivity is contained inside the containment building. The acceptance criteria are designed to check that a normal containment environment exists or that the operator is alerted to an off-normal condition. Containment pressure greater than the maximum expected normal containment pressure, high radiation inside or outside containment, or the steam plant are indications that more than an uncomplicated reactor trip has occurred.

CONTINGENCY ACTIONS

If a steam plant radiation monitor is in alarm, steps are provided to secure feedwater to the most affect steam generator. Since the steam generator level rises due to the leakage from the RCS, adding additional makeup could result in a potential overfill.

Contingency actions are designed to ensure that the containment is isolated when containment pressure reaches the CIAS setpoint. Additionally actuation for SIAS, EBFAS, and MSI are checked, along with the CIAS actuation. Once complete facility of Control Room Air Conditioning (CRACS) is also checked in service. The CRACS is checked to satisfy the Control Room Habitability Analysis.

JUSTIFICATION FOR DEVIATION

The EPG checks containment pressure first, then radiation monitors. MP2 checks the radiation monitors first, then containment pressure. The Containment Temperature and Pressure Control and Containment Combustible Gas Control Safety Functions both require the Unit Supervisor to obtain containment pressure. By placing containment pressure last in this Safety Function, it allows the Unit Supervisor to proceed through the next two Safety Functions without having to query the PPO for this information.

MP2 adds a step to check radiation monitors outside containment. This would alert the operator of a LOCA occurring outside containment. This is also consistent with the guidance in the EPG for a LOCA.

MP2 adds a Contingency Action to throttle/isolate feedwater to the most affected steam generator following a SGTR. This action is necessary to prevent a possible overfill of the ruptured steam generator.

SRO Exam Questions Only (No "Parents" Or "Originals")

Question #: 5	Question ID: 9082581	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
I-SRO Ques. # 5	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

The plant has experienced a loss of VA-10 while in Mode 5 with Shutdown Cooling in operation.

Assuming RBCCW flow was NOT diverted from the SDC Heat Exchangers by any other system, which of the following actions would be performed outside the Control Room and what is the reason for performing these actions in the listed order?

☐ **A** 1. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.
2. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.

2-SI-306, SDC Total Flow Control Valve, must be opened first to provide minimum flow for the operating LPSI Pump.

☑ **B** 1. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.
2. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.

2-SI-657, SDC Heat Exchanger Flow Control Valve, must be opened first to establish the desired RCS cooldown rate.

☐ **C** 1. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.
2. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.

2-SI-657, SDC Heat Exchanger Flow Control Valve, must be opened first to establish the desired RCS cooldown rate.

☐ **D** 1. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.
2. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.

2-SI-306, SDC Total Flow Control Valve, must be opened first to provide minimum flow for the operating LPSI Pump.

Justification

B IS CORRECT; AOP 2572, Loss of SDC, requires the PEO to first place SI-657 in local manual control to restore cooling to the RCS. SI-657 is a reverse operating valve (i.e., clockwise rotation is open, counterclockwise is close). On a loss of power (VA-10), SI-657 fails closed; therefore, SI-657 must be rotated in the *clockwise* direction to open it. SI-306 is also a reverse operated valve. SI-306 fails open on a loss of power (VA-10), and must be rotated in the *counterclockwise* direction to throttle it closed. SI-306 is operated last to allow more flow through the SDC Heat Exchangers, if necessary to provide additional cooling flow to the RCS.

A is incorrect; The order of local valve operations is incorrect. Although SI-657 fails closed, SI-306 fails open; therefore, there is no need to establish minimum flow protection for the running LPSI Pump.
Plausible because the examinee may think that it is more important to initiate flow through the heat exchanger bypass (Total flow Control valve) than to initiate flow through the SDC Heat Exchanger. This may allow for a more controlled initiation of the cooldown.

C is incorrect; The order of valve operations is correct; however, the direction of valve rotation is incorrect. Both valves are reverse operating.
Plausible because the examinee may not remember that both valves are reverse operated.

D is incorrect. The order of local valve operations is incorrect and the direction of valve rotation is incorrect.
Plausible because the examinee may think that this sequence allows for more control of the cooldown. Additionally, the examinee may not remember that both of these valves are reverse operated.

References

AOP 2572 Loss of SDC, section 8.0

Comments and Question Modification History

Bob K. - D-3/W (memory on reverse acting)
Bill M. - D-3/C, 50/50
Angelo - D-3/C; No comments.

Question #: **5** | ***Question ID:*** ***9082581*** | ☐ ***RO*** | ☑ ***SRO*** | ☐ **Student Handout?** | ☑ **Lower Order?**

I-SRO Ques. # **5** | **Rev. 0** | ☑ **Selected for Exam** | ***Origin:*** ***New*** | ☐ **Past NRC Exam?**

NRC K/A System/E/A **System** 057 Loss of Vital AC Electrical Instrument Bus

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.35 **RO** 3.8 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Millstone Unit 2 Loss of Shutdown Cooling	AOP 2572 Revision 009-04 Page 45 of 70

8.0 Loss of Power or Air to SI-657, SI-306 or Both

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

1. Loss of power or air to SDC flow control valves has the following affect:
   - SI-657 fails closed
   - SI-306 fails open to its limit stop (mid-position)
2. Loss of VA-10 fails both valves.
3. SI-306 can be throttled from its maximum open position (maximum flow limit stop position) when diverting additional flow through SDC heat exchangers is required.
4. Obtaining reference positions of SDC flow control valves may be helpful during SDC restoration. If a loss of VA-10 has occurred, the reference positions are only available as archive data in the PPC (PPC analog points 2SI657 and 2SI306).

___8.1 OBSERVE applicable controllers or PPC analog points to obtain a reference position for SDC flow control valves:

- Output of FIC-306 or archive PPC data for 2SI306
- Output of HIC-3657 or archive PPC data for 2SI657

___8.2 For the failed valve(s), ADJUST the controller output to match actual valve position.

- FIC-306
- HIC-3657

Question #: 5 | Question ID: 9082581 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

I-SRO Ques. # 5 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Millstone Unit 2 Loss of Shutdown Cooling	AOP 2572 Revision 009 – 04 Page 49 of 70

INSTRUCTIONS | CONTINGENCY ACTIONS

CAUTION

Care should be used when establishing SDC heat exchanger flow due to the potential for water in the SDC heat exchanger to be much cooler than RCS temperature. Initiating flow slowly allows temperatures to equalize.

NOTE

1. SI-657 is a reverse-operating valve; *counterclockwise* rotation of the handwheel closes the valve and *clockwise* rotation of the handwheel opens the valve.
2. When establishing RCS cooldown rate, optimum temperature response is achieved by maintaining SI-657 between 35 and 60% open.

___8.8 IF SI-657 had a loss of power or air, PERFORM the following:

a. WHEN establishing cooldown, Refer to SP 2602B, "Transient Temperature, Pressure Verification," and PERFORM the following:

- MONITOR RCS cooldown rate using T351Y.
- ENSURE system response is within cooldown limits.

b. PERFORM the following to take manual control of SI-657 ("A" ESF Room):

1) UNLOCK the manual handwheel on valve.

2) CLOSE instrument air supply valve and VENT valve operator.

3) LOOSEN stem hex nut as required to allow stem movement.

4) ROTATE SI-657 handwheel as directed by the Control Room.

(continue)

Question #: 5 | Question ID: 9082581 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?
I-SRO Ques. # 5 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Millstone Unit 2 Loss of Shutdown Cooling	AOP 2572 Revision 009 – 04 Page 52 of 70

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

SI-306 is a reverse-operating valve; *counterclockwise* rotation of the handwheel closes the valve and *clockwise* rotation of the handwheel opens the valve.

___8.9 As necessary, ESTABLISH local manual control of SI-306 as follows:

a. ESTABLISH communications between operators at the valve ("A" ESF Room) and Control Room.

b. CLOSE instrument air supply for SI-306, SDC total flow control.

c. OPEN petcock on instrument air supply pressure regulator and VENT SI-306, SDC total flow control, valve operator.

d. UNLOCK and REMOVE chain from manual handwheel.

e. ROTATE manual handwheel *counterclockwise* and ALIGN holes in outer shaft with hole in inner shaft.

f. INSERT the pin into shaft holes.

g. ENSURE SI-306, SDC total flow control, valve position indicator on the manual actuator is at throttled open position.

h. IF desired to manually operate valve, POSITION SI-306 handwheel as directed by the Control Room.

___8.10 WHEN recovery from manual operations is desired, PERFORM actions specified by the SM/US.

Question #: **6** | ***Question ID:*** ***3100002*** ☐ **RO** ☑ **SRO** ☐ **Student Handout?** ☑ **Lower Order?**

I-SRO Ques. # 6 | Rev. 2 ☑ **Selected for Exam** ***Origin:*** ***Bank*** ☐ **Past NRC Exam?**

The plant is operating at 100% power when the Balance of Plant (BOP) operator reports that Instrument Air header pressure is at 95 psig and lowering. Immediately following, the Turbine Building PEO reports a large unisolable leak just downstream of the "D" Instrument Air Dryer After Filters.

Assuming Instrument Air header pressure continues to lower, at what pressure in the Instrument Air System must the Unit Supervisor (US) direct a manual reactor trip (by procedure) and why?

☐ **A** Prior to reaching 85 psig.
When pressure drops below 85 psig the crew is procedurally directed to crosstie Station Air with Unit 3. Operation in this alignment will result in all components supplied by Instrument Air being inoperable, which is an unanalyzed condition.

☐ **B** When pressure lowers to less than 85 psig.
At approximatley 85 psig the Instrument Air/Station Air Crosstie valve opens. Continued operation with Station Air supplied to valves and controllers will result in erratic operation of components due to the high moisture content of Station Air.

☑ **C** When pressure lowers to less than 80 psig.
The loss of many important controls, such as Feedwater, could degrade plant conditions at the time of the trip; therefore, the reactor must be tripped when control of important systems could become challenged.

☐ **D** Prior to reaching 80 psig.
The Auxiliary Feed Regulating Valves will lock up with less than 80 psig supply pressure. The reactor must be tripped to allow the initial automatic opening of these valves and begin feeding Steam Generators.

Justification

C IS CORRECT; AOP 2563, Discussion section 1.2. When IA pressure lowers to less than 80 psig, the Feed Regulating Valves may lock up resulting in over feeding of Steam Generators after the trip. Additionally, the Steam Dumps may not open resulting in opening of the Main Steam Safeties as the only initial means of removing decay heat.

A is incorrect; Although it is less desirable to operate with Unit 2 cross tied with Unit 3, there are NO restrictions; therefore, NO requirements to trip.
Plausible because the examinee may feel that continued operation with Station Air supplied by Unit 3 (and, subsequently Station Air crosstied to Instrument Air) is NOT allowed.

B is incorrect; Although the Instrument Air/Station Air Crosstie valve automatically opens at ~ 85 psig, continued operation with Station Air cross tied to Instrument Air is acceptable.
Plausible because the examinee may remember that the Station air Cross Tie valve opens, he/she may think that continued operation with Station Air supplying Instrument Air is NOT allowed.

D is incorrect; The Auxiliary Feed Regulating Valves have back up air that will ensure their operation for a limited duration even during a complete loss of Instrument Air.
Plausible because the examinee may feel that the potential loss of Auxiliary Feedwater requires an immediate Reactor trip.

References

AOP 2563, Loss of Instrument Air, Section 1.2

Comments and Question Modification History

Bob K. - D-2/C (only need to know "80#")
Changed pressure values on "A", "B" and "D" and reworded "A" and "B" slightly. - RLC
Bill M. - D-2/C, K (90 psig is too high.)
Changed values in distractors "A" and "B" to 85 psig vs 90 psig. More plausible. - RJA
Angelo - D-3/C; No comments.

NRC K/A System/E/A **System** 065 Loss of Instrument Air

Number AA2.06 **RO** 3.6* **SRO** 4.2 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is decreasing

1.0 PURPOSE

1.1 Objective

This procedure provides the operator with specific steps to be taken following a significant drop in instrument air pressure and actions to mitigate the effects of a reactor trip when the Instrument Air System is *not* available.

1.2 Discussion

This procedure is implemented when instrument air pressure is lowering below normal values.

With a complete loss of instrument air, continued steady state plant operation is *not* possible. The loss of many important controls, such as in the feedwater system, could degrade plant conditions at the time of a reactor or turbine trip. Therefore, the reactor is tripped immediately when instrument air pressure lowers to the point where control of important systems is questionable. This may be indicated by system response or instrument air header pressure of less than 80 psig.

Should instrument air header pressure drop suddenly, as in the case of a main header rupture, the only initial means of decay and sensible heat removal is the main steam safeties. Subsequently, manual control of the atmospheric dump valves and use of the Auxiliary Feedwater System can mitigate the pressure transient, thereby removing sensible and decay heat from the RCS in a controlled manner.

Question #: 7 | **Question ID: 9000020** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 7 | Rev. 0 | ☑ Selected for Exam | **Origin: New** | ☐ Past NRC Exam?

The reactor is at 100% power with the CEA Motion surveillance in progress. When Group 7 CEA #1 is tested, CEAPDS indicates it inserts two steps, then slips an additional 20 steps. The appropriate actions were taken to stabilize RCS temperature and the following conditions were observed:

* Reactor power stable at ~ 99%.
* <u>Only</u> Upper Electrical Limit lights are energized on the core mimic.
* CEA #1 indicates 158 steps withdrawn on CEAPDS.
* CEA #1 indicates 178 steps withdrawn on the PPC.
* CEA Motion Inhibit (CMI) alarms on C-04
* CEAPDS Group Deviation indication for CEA #1

Fifty (50) minutes after CEA #1 slipped, all required actions per AOP-2556, CEDS Malfunctions, have been completed, including plant power changes.
Also, I&C reports the circuit malfunction that caused CEA #1 to slip has been repaired and the CEA can now be recovered.

Which one of the following describes actions that must be taken to recover CEA #1 <u>and</u> what is the administrative concern of those actions?

☐ **A** Pulse counts must be reset to clear the Upper Core Stop <u>and</u> the CMI must be bypassed for CEA recovery.
CEA #1 Pulse Count Indication <u>and</u> the CMI will be INOPERABLE while the CEA is being recovered.

☐ **B** CEA #1 Upper Electrical Limit must be overridden <u>and</u> the CMI must be bypassed for CEA recovery.
Reed Switch Indication for CEA #1 <u>and</u> the CMI will be INOPERABLE while the CEA is being recovered.

☐ **C** Pulse counts must be reset to clear the Upper Core Stop <u>and</u> the CMI must be bypassed for CEA recovery.
<u>Only</u> the CMI will be INOPERABLE while the CEA is being recovered.

☑ **D** CEA #1 Upper Electrical Limit must be overridden <u>and</u> the CMI must be bypassed for CEA recovery.
<u>Only</u> the CMI will be INOPERABLE while the CEA is being recovered.

Justification

D - CORRECT; CMI is triggered based on the CEA #1 deviation from the other CEAs in Group 7, therefore it must be bypassed to recover the CEA. When the CMI is bypassed, it is considered INOPERABLE. Also, the UEL reed switch indicates it is stuck on. This failure is <u>not</u> considered a failure of the CEA Indication System, but does require additional action be taken to recover the CEA.

A - WRONG; CEA pulse counting indication is no longer accurate for CEA #1, so it is reset to the actual slipped rod position (based on reed switches) before CEA recovery is attempted. This does not make the pulse counting indication INOPERABLE.
<u>Plausible</u>; Examinee may think the reed switch input to the Core Mimic also triggers the Upper Core Stop on the PPC, because Core Mimic reed switches reset the PPC pulse counts and the UCS is armed under the existing conditions.

B - WRONG; These are the correct interlocks/inhibits that must be bypass to recover the CEA. However, bypassing the reed switch input to the UEL does not make it INOPERABLE.
<u>Plausible</u>; Examinee may think bypassing UEL interlock has same impact as bypassing CMI interlock, especially where I&C "must lift a lead" in the field to bypass the UEL interlock.

C - WRONG; The CMI and the UEL interlock must be bypassed to withdraw the slipped CEA.
<u>Plausible</u>; Examinee may think bypassing the CMI, which is based on reed switch input, bypasses all interlocks based on reed switch input (including the reed switches that drive the UEL).

References

AOP 2556, Pages 3, 13 and 14

Comments and Question Modification History

Bob K. - D-4/C (Add to stem that all required actions of AOP-2556 have been completed up to CEA recovery.)
Reworded stem per comment - RLC
Bill M. - D-3/C, 50/50 (Need to reword last bullet. Meaning is not clear.)
Split last bullet to specify alarms on C-04 and CEAPDS indication for CEA #1. - RJA
Angelo - D-5/W; Difficult but fair.

Question #: 7 | Question ID: 9000020 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 7 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

NRC K/A System/E/A System 003 Dropped Control Rod

Number AA2.04 **RO** 3.4* **SRO** 3.6* **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod motion stops due to dropped rod

Millstone Unit 2 CEA Malfunctions	AOP 2556 Revision 016–10 Page 3 of 55

1.0 PURPOSE

☞ This AOP contains EOP related material. ☜

1.1 Objective

This procedure provides instructions for the following malfunctions which could affect CEAs, CEDS, ACTM or CEA position indications:

- Multiple misaligned or untrippable CEAs
- Misaligned CEA misaligned greater than 10 steps
- Inoperable CEA Position Indication System
- Inoperable CMI circuit
- Trippable CEA
- Untrippable CEA

1.2 Discussion

Following a CEA drop, operator action should be directed toward returning the plant to a stable condition. At high power levels, if no action is taken following the CEA drop, reactor power will return to approximately the initial power level, but at a reduced core average temperature (due to positive reactivity feedback from the negative moderator temperature coefficient). The following actions will minimize the affects of the CEA drop transient.

- Dropped CEAs result in reactor and turbine power mismatch. This mismatch is nulled by reducing turbine load to match reactor power (RCS temperatures *not* changing).
- Dropped CEAs also result in undesirable neutron flux patterns. By correct operator response, the time span over which these patterns exist is minimized.
- It is desirable to record as much data as possible concerning abnormal flux patterns existing during and subsequent to rod drop. Proper use of PPC, as outlined in this procedure, produces data necessary for subsequent analysis by Reactor Engineering.

If during CEA motion, a core mimic light fails to clear or light and all other indications are normal, it is acceptable to request I&C Department to temporarily lift leads for respective CEA upper electrical limit reed switch until it is aligned. This is *not* considered a CEA position indication problem.

Level of Use Continuous   STOP THINK ACT REVIEW

Millstone Unit 2 CEA Malfunctions	AOP 2556 Revision 016-10 Page 13 of 55

INSTRUCTIONS	CONTINGENCY ACTIONS
___4.15 IF necessary, PERFORM the following to modify dropped CEA position on PPC to match dropped CEA position indicated on "CEAPDS MONITOR:" a. OBSERVE dropped CEA position on "CEAPDS MONITOR." b. Using "CEA Positions Menu" on PPC, SELECT "CEA POSITION EDITOR" and PERFORM directions as indicated on PPC.	

Question #: 7 | Question ID: 9000020 | ☐ RO ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
I-SRO Ques. # 7 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Millstone Unit 2 CEA Malfunctions	AOP 2556 Revision 016 – 10 Page 14 of 55

INSTRUCTIONS	CONTINGENCY ACTIONS
___4.19 As desired, SELECT applicable CEA group " +/- 15 STEPS" or "FULL SCALE" on "CEAPDS MONITOR."	
___4.20 PRESS "MANUAL INDIVIDUAL, MI" and CHECK light lit.	
___4.21 PRESS applicable *group* "INHIBIT BYPASS" and CHECK the following: • Appropriate group red "INHIBIT BYPASS," lit • "CEA MOTION INHIBIT BYP" annunciator lit (BA-19, C-04) (depends on group selected)	
___4.22 PRESS applicable "GROUP SELECTION."	
___4.23 LOG entry into T/S LCO, 3.1.3.1, ACTION B.1 (CMI bypassed).	

Question #: **8** | **Question ID: 9000005** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

I-SRO Ques. # **8** | **Rev. 0** | ☑ **Selected for Exam** | **Origin: New** | ☐ **Past NRC Exam?**

The reactor is manually tripped from 100% power due to a Steam Generator Tube Rupture (SGTR). On the trip, the RSST is lost due to grid instabilities. All other systems respond normally. EOP 2525, Standard Post Trip Actions, is entered.

What is the direction for controlling the affected SG level during the performance of EOP 2525, Standard Post Trip Actions, and what is the basis for this direction?

☐ **A** Secure all feedwater to the affected SG. This will provide the maximum volume to accept water from the tube rupture and still allow a cooldown to 515°F to isolate the affected SG.

☐ **B** Maintain the affected SG level 40 to 70%. This will maintain the SG tube covered to allow a cooldown to 515°F and still maintain adequate volume to accept water from the tube rupture.

☐ **C** Maintain at least 300 gpm feedwater to the affected SG for Heat Removal. The addition of clean water will provide dilution of radioactivity which will lower the release to the environment.

☑ **D** Feed the affected SG to maintain level 40 to 45%. This will cover the SG tube to allow for Iodine scrubbing and still allow adequate volume to accept water from the tube rupture.

Justification

D IS CORRECT; If Steam Plant Radiation Monitors are in alarm, EOP 2525 requires feed flow to maintained to the SG with the highest radiation reading to maintain level 40 to 45%. The lower limit of 40% is above the top of the tubes. Keeping the tubes covered allows for Iodine scrubbing and limits the gaseous release to the environment (The loss of off-site power results in a loss of condenser vacuum which requires the use of the ADVs for heat removal control). The upper limit of 45% allows for a significant volume to accept water from the RCS through the broken tube(s).

A is incorrect; Feedwater should NOT be secured to the affected SG until after the cooldown to 515°F. The addition of feedwater allows for Iodine scrubbing and limits the radioactive release to the environment.
Plausible because, up until a few years ago, feedwater was secured to the affected SG immediately after a SGTR was diagnosed.

B is incorrect. If affected SG level is maintained higher than 45%, then subsequent leakage into the SG from the RCS may result in overfilling the SG and could ultimately result in radioactive water being discharged out the affected SG ADV to the environment.
Plausible because the normal post trip SG level is 40 to 75%.

C is incorrect. While it is true that the addition of feedwater will provide additional Iodine scrubbing and dilution of radioactive contaminants, too much feedwater flow will result in overfilling the SG and could ultimately result in radioactive water being discharged out the affected SG ADV to the environment.
Plausible because EOP 2525 requires feeding the unaffected (or least affected) SG at greater than 300 gpm.

References

OP 2260, Unit 2 EOP User Guide
EOP 2525, Standard post Trip Actions

Comments and Question Modification History

Bob K. - D-2/C (need only "40% - 45%" for answer)
No changes made.
Bill M. - D-2/C, K
Angelo - D-2/C; No comments.

NRC K/A System/E/A **System** 060 Accidental Gaseous Radwaste Release

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.18 **RO** 3.3 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.1 / 45.13)
Knowledge of the specific bases for EOPs.

Attachment 1
EOP 2525, "Standard Post Trip Actions," Implementation Guide
(Sheet 8 of 11)

b. Due to current MSSV blowdown setpoints (~880 psia), it is possible for MSSV(s) to be open post–trip when the steam dumps are not available AND have SG pressure within the normal control band. To assess if a MSSV is actually stuck open, it is recommended the CO adjust ADV automatic setpoint(s) to the lower end of the control band. Additionally, temperature outside the normal band should also be used to assess a stuck open MSSV.

c. If T_C is less than 530° F, the operator should determine if feedwater flow is excessive to one or both SGs and adjust or isolate feedwater flow as required. The operator should determine if the cause of the excessive flow is due to control system malfunction (e.g., FRV not closing on the trip or FRV bypass valves not at the required position following the trip), or due to feedwater flow to a SG blowing down from an ESDE.

- In the case of malfunctioning equipment, the operator should attempt to adjust feedwater flow manually, since a component failure should not result in the loss of a SG for heat removal.

- The operator is required to isolate AFW to the affected SG within 30 minutes following the generation of MSIS during an ESDE. For scenarios where isolation is not possible from the Control Room, allowance must be made for local operation of FW–43A(B) or FW–44. It has been validated that it will take approximately 15 minutes to close FW–43A(B) or FW–44 locally. Therefore, isolation of AFW to an affected steam generator, from the control room, must be attempted within 15 minutes of a MSIS.

d. If T_C is less than 530° F AND the ESDE has been terminated, the operator is required to operate the ADV or steam dumps to stabilize T_C. Temperature should not be allowed to restore to the normal band following an ESDE.

e. If SG level is lowering and both MDAFW pumps are not operating, the operator is required to start the TDAFP within 10 minutes following a Loss of Normal Feedwater.

f. If a SGTR has occurred, the operator is expected to throttle feed to the most affected SG as necessary to maintain level low in the band (40 to 45%). This will aid in maintaining SG pressure during the cooldown and aid in scrubbing radioactive iodine. The top of the SG tube bundle is ~33%. If break flow is restoring level to this band then feed flow is not necessary [this may not be assessed until verification of Containment Isolation].

SRO Exam Questions Only (No "Parents" Or "Originals")

Question #: 9	**Question ID:** 9000006	☐ RO	☑ SRO	☑ Student Handout?		☐ Lower Order?	
I-SRO Ques. # 9	**Rev.** 0	☑ Selected for Exam		**Origin:** New		☐ Past NRC Exam?	

The plant is in MODE 5 performing a normal cooldown for refueling. "B" LPSI Pump is in service supplying both SDC Heat Exchangers. RCS to SDC Temperature, T351X, is presently reading 187°F with RCS pressure being held at 150 psia.

Suddenly, Bus 24D is deenergized due to a fault. Fifteen minutes after the loss of Bus 24D, the following conditions are reported:

- RCS pressure is 155 psia and slowly rising.
- RCS to SDC Temperature, T351X, is reading 186°F and stable.
- CET temperatures are 205°F and slowly rising.
- RVLMS indicates vessel level at 100%.
- Both S/G levels are 60% and stable
- Containment is being evacuated.

NO other operator actions have been taken.

Which of the following notifications must be made?

☐ **A** General Interest, Echo

☐ **B** Unusual Event, Delta-One

☑ **C** Alert, Charlie-One

☐ **D** Site Area Emergency, Charlie-Two

Justification

C IS CORRECT; MP-26-EPI-FAP06-002, Equipment Failure, EA2, Uncontrolled temperature increase >10°F that results in RCS temperature >200°F. Due to the loss of SDC flow, T351X, is no longer providing an accurate RCS temperature. The operator must use CET temperatures to determine the actual change in RCS temperature.

A is incorrect; The loss of Bus 24D would result in an Undervoltage actuation on Facility 2. Per RAC 14, Non-Emergency Station Events, an 8 hour report is required. (General Interest, Echo); however, the loss of SDC with a temperature rise of >10°F is a higher classification. The highest classification must be reported. Other details may be included in the initial report.
Plausible if examinee thought that this was the only reportable event.

B is incorrect. Per MP-26-EPI-FAP06-002, Equipment Failure, EU1(2.), Uncontrolled temperature increase >10°F, a classification of Unusual Event Delta-One may be reported; however, the loss of SDC with a temperature rise of >10°F is a higher classification. The highest classification must be reported.
Plausible if examinee did not look at the higher Alert classification for Inability to Maintain Cold S/D.

D is incorrect; Per MP-26-EPI-FAP06-002, Equipment Failure, ES2(1.), No RCS Heat Removal via Steam Generators AND Once Through Cooling NOT effective AND Shutdown Cooling NOT in service. This may appear to be a logical choice in that none of the 3 core cooling methods are being utilized; however, Steam Generators are available for Heat Removal when RCS temperature is high enough to cause steaming AND Once Through Cooling would likely be effective if it were initiated.
Plausible because the examinee may believe that no core cooling method is presently being utilized, therefore, core cooling is in serious jeopardy.

References **Provided**

MP-26-EPI-FAP06-002, Millstone Unit 2 Emergency Action Levels, and RAC-14, Non-Emergency Station Events

Comments and Question Modification History

Bob K. - D-3/C
Bill M. - D-2/Y, K (Add, "No other operator actions have been taken." to the bullet list)
Added "No other operator actions have been taken." to the bullet list, per recommendation. RJA
Angelo - D-3/C; No comments.

NRC K/A System/E/A	**System**	074	Inadequate Core Cooling
Generic K/A Selected			
NRC K/A Generic	**System**	2.4	Emergency Procedures /Plan

Question #: **9** | **Question ID: 9000006** | ☐ RO | ☑ SRO | ☑ Student Handout? | ☐ Lower Order?

I-SRO Ques. # **9** | **Rev. 0** | ☑ Selected for Exam | **Origin: New** | ☐ Past NRC Exam?

Number 2.4.30 **RO** 2.7 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 43.5 / 45.11)

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

EQUIPMENT FAILURE

EG1	ATWS/INADEQUATE COOLING	Mode 1

ES1	ATWS	Mode 1

ES2	INABILITY TO MAINTAIN HOT S/D	Mode 1, 2, 3, 4

ES3	IN-VESSEL FUEL UNCOVERY	Mode 5, 6

ES4	LOSS OF ANNUNCIATORS/TRANSIENT	Mode 1, 2, 3, 4

EA1	AUTOMATIC RX TRIP FAILURE	Mode 1, 2

Failure of Automatic Reactor Trip AND Manual Trip Was Successful

EA2	INABILITY TO MAINTAIN COLD S/D	Mode 5, 6

1. Uncontrolled RCS Temperature Increase > 10°F That Results In RCS Temperature > 200°F
2. Inadvertent Criticality

EA3	LOSS OF ANNUNCIATORS/TRANSIENT	Mode 1, 2, 3, 4

Loss of Most (75%) MCB Annunciators > 15 Minutes AND EITHER of the Following:

- Significant Transient In Progress
- Loss of SPDS AND ICC Instrumentation

EU1	LOSS OF COLD S/D FUNCTION	Mode 5, 6

1. Loss of Shutdown Cooling > 15 Minutes AND Refuel Pool Water Level < 35 Ft., 6 In.
2. Uncontrolled RCS Temperature Increase > 10°F
3. RCS Boron Concentration < Minimum Required

EU2	REFUEL/SPENT FUEL POOL LEVEL	Mode 6, 0

1. Uncontrolled Spent Fuel Pool Water Level Decrease Causing Loss of Cooling Suction Flow
2. Uncontrolled Refuel Pool Water Level Decrease Requiring Containment Evacuation AND All Spent Fuel Assemblies in Safe Storage Locations

EU3	LOSS OF ANNUNCIATORS	Mode 1, 2, 3, 4

Loss of Most (75%) MCB Annunciators > 15 Minutes AND SPDS OR ICC Instrumentation Available

EU4	LOSS OF COMMUNICATIONS	Mode ALL

1. Loss of ALL Onsite Electronic Communications Methods
2. Loss of ALL Electronic Communications Methods With Government Agencies

EU5	SHUTDOWN LCO EXCEEDED	Mode 1, 2, 3, 4

Unit NOT Brought To Required Mode Within Applicable LCO Action Statement Time Limits

Question #:	10	Question ID:	9000007	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
I-SRO Ques. #	10	Rev.	0	☑ Selected for Exam	Origin: New		☐ Past NRC Exam?

An RCS chemistry sample taken at 100% power indicates 6 micro-curies/gram DOSE EQUIVALENT I-131.

Which of the following describes the required action and the basis for that action?

☐ **A** With the specific activity of the primary coolant > 0.1 micro-curies/gram DOSE EQUIVALENT I-131, be in COLD SHUTDOWN within 36 hours after detection. Isotopic analysis of the primary coolant must be performed once per hour when activity of the primary coolant > 0.1 micro-curies/gram DOSE EQUIVALENT I-131.
The hourly sampling period allows time to obtain and analyze a sample. There is a low probability of a steam line break or S/G tube rupture in the next 36 hours and there is significant conservatism built into the RCS specific activity limit.

☑ **B** With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 ≤ 60 micro-curies/gram once per 4 hours. Operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 micro-curies/gram limit.
The 4 hour sampling period allows time to obtain and analyze a sample. There is a low probability of a steam line break or S/G tube rupture in the next 48 hours and it is expected that normal coolant Iodine concentration would be restored within 48 hours.

☐ **C** With the specific activity of the primary coolant > 0.1 micro-curies/gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours from the time of detection and in COLD SHUTDOWN within 48 hours from the time of detection.
The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and prevent exceeding the radiological release limit at the site boundary from an assumed LOCA.

☐ **D** With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, lower the RCS specific activity to ≤ 1.0 micro-curies/gram DOSE EQUIVALENT I-131 within the next 36 hours or be in HOT STANDBY within the following 6 hours.
It is expected that normal coolant Iodine concentration would be restored within 36 hours. If not, adequate time is provided to achieve HOT STANDBY to prevent exceeding Control Room dose limits from an assumed LOCA.

Justification

B is CORRECT; TSAS 3.4.8a. and b. state: a. With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 ≤ 60 micro-curies/gram once per 4 hours. b. With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, but ≤ 60 micro-curies/gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 micro-curies/gram limit. The Basis for TS 3.4.8 states: With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ 60 micro-curies/gram. Four hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend. The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were an Iodine spike, normal coolant Iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A is incorrect; TSAS 3.4.8 does not require the plant to achieve COLD SHUTDOWN within 36 hours.
Plausible because several other Tech Spec Action Statements require the plant to achieve COLD SHUTDOWN within 36 hours (Example: Containment Integrity, TSAS 3.6.1.1).

C is incorrect; TSAS 3.4.8 does not require the plant to achieve HOT SHUTDOWN within 6 hours.
Plausible because several other Tech Spec Action Statements require the plant to achieve HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 36 hours. (Example: Specific Activity, TSAS 3.4.8c., DOSE EQUIVALENT I-131 >60 micro-curies/gram.)

D is incorrect; TSAS 3.4.8 does not require the plant to achieve HOT SHUTDOWN within 6 hours if RCS coolant specific activity cannot be lowered to ≤ 1.0 micro-curies/gram DOSE EQUIVALENT I-131 in 36 hours.
Plausible if the examinee confuses the time requirements and the actual limit.

References

Tech Spec 3.4.8, Specific Activity, and applicable Bases.

Comments and Question Modification History

Bob K. - D-4/W (Could not remember requirement details. Did not believe should be required from memory.)
Changed distractors A and C to 0.1 micro-curies/gram instead of the original 1.0 micro-curies/gram. The different limits require the examinee to remember the actual limit and not just the action required if the limit is exceeded. RJA

Question #: **10** | **Question ID: 9000007** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

I-SRO Ques. # 10 | **Rev. 0** | ☑ **Selected for Exam** | **Origin: New** | ☐ **Past NRC Exam?**

Bill M. - D-4/W, G (Could not remember requirement details. Examinees should get this if they studied Tech Specs and Bases.)
Angelo - D-5/C; Difficult but fair. **Changed 1.0 in "A" to 0.1 to match number in first sentence. - RLC**

NRC K/A System/E/A **System** 076 High Reactor Coolant Activity

Number AA2.02 **RO** 2.8 **SRO** 3.4 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS

Question #:	10	Question ID:	9000007	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?
I-SRO Ques. #	10	Rev.	0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

October 27, 2008

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 1100 μCi/gram DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, 4.

ACTION:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gram once per 4 hours.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but ≤ 60 μCi/gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 μCi/gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or > 60 μCi/gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within 36 hours.

d. With the specific activity of the primary coolant > 1100 μCi/gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 1100 μCi/gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the primary coolant > 1100 μCi/gram DOSE EQUIVALENT XE-133 for more than 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within 36 hours.

Question #: **11** | ***Question ID:* 9000008** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☐ **Lower Order?**

I-SRO Ques. # 11 | **Rev. 0** | ☑ **Selected for Exam** | ***Origin:* New** | ☐ **Past NRC Exam?**

A plant heatup has just been started and the following conditions presently exist:

- RCS Temperature is at 205°F and slowly rising.
- RCS pressure is stable at the minimum allowed for "A" and "B" RCP operation.
- "A" and "B" RCPs have just been started.
- Shutdown Cooling has just been secured.
- "C" and "D" RCP breakers have just been racked up.

Then, the "B" RCP trips when the breaker's overcurrent relay actuated due to being jarred while moving staging (NOT an actual overcurrent condition).

Which of the following actions are required under the present conditions?

☑ **A** Immediately secure the "A" RCP, raise RCS pressure, then start "C" and "D" RCPs.

☐ **B** Immediately place Shutdown Cooling back in operation, then secure the "A" RCP.

☐ **C** Immediately start "C" and "D" RCPs, then secure the "A" RCP.

☐ **D** Immediately start the "C" RCP and operate it with the "A" RCP.

Justification

A - CORRECT: The NPSH required pressure for "A" & "B" RCPs is based on both pumps operating, therefore, the "A" RCP must be immediately secured. With a plant heatup in operation, OP-2201 states RCS pressure should be raised as required to allow for the start of the available RCPs, then they should be started.

B - WRONG; If NPSH requirements are not met, the RCP should be immediately secured.
Plausible; Would be chosen if avoiding the loss all RCS flow (violation of Tech. Specs.) is considered above possible RCP damage.

C - WRONG; The NPSH required pressure for "A" & "B" RCPs is less than that for "C" & "D" RCPs (see OP-2201, Attach. 2 & 3). If the RCS pressure s at the minimum allowed for "A" & "B" RCP operation (initial conditions), it must be below the minimum required pressure for "C" & "D" RCP operation.
Plausible; Would be chosen if "C" & "D" RCP NPSH required pressure is believed to be lower than "A" & "B" RCPs.

D - WRONG; Two pump operation is specific to the applicable pumps as one pump aids in the NPSH requirements of the other.
Plausible; Would be chosen if two pump operation is known, but believed to be any two RCPs operating simultaneously.

References

OP-2301, Pg. 24; Caution and Pg. 68; Attach. 6, Conditional Actions.

Comments and Question Modification History

Bob K. - D-4/W (Reword to make clear "relay failure" caused breaker to trip and is a quick fix).
Reworded stem to clairify problem is with the breaker's overload relay and is a relatively quick fix. - RLC
Bill M. - D-3/C, K
Angelo - D-2/W; **Changed RCS temperature to 205°F to allow all four RCP breakers to be closed. - RLC**

NRC K/A System/E/A **System** 003 Reactor Coolant Pump System (RCPS)

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.23 **RO** 4.3 **SRO** 4.4 **CFR Link** (CFR: 41.10 / 43.5 / 45.2 / 45.6)
Ability to perform specific system and integrated plant procedures during all modes of plant operation.

CAUTION

1. Two RCPs in the same loop are started to ensure proper NPSH for the pumps. To ensure proper NPSH, the second pump should be started *immediately* after first pump starting current has decayed.
2. Upon RCP start the initiation of bypass spray flow and any potential in–surge, may result in a lowering of RCS pressure. Prompt operation of pressurizer heaters (proportional and backup), should be anticipated.

NOTE

1. When starting RCPs, it is desirable to maintain RCS pressure between RCP MNPSH and 265 psia, as indicated on Attachment 2, 3 or the applicable PPC NPSH display, to accommodate any pressure change on start.
2. If a temperature rise is expected upon RCP start, this must also be considered part of the RCS heatup.

4.4.12 After starting RCPs, perform the following:

___/___ • SDC to RCS temperature, T351Y, must be lowered rapidly to compensate for the extra heat input from the RCPs.

___/___ • SDC to RCS temperature, T351Y, is then raised to allow heatup to progress.

___/___ • Monitor pressurizer pressure.

___/___ 4.4.13 Refer To OP 2301C, "Reactor Coolant Pump Operation" and PERFORM applicable actions to start selected RCPs (C–03).

___/___ 4.4.14 WHEN RCPs are operating, Refer To Attachments 1 and 2 or 3, or applicable PPC displays, and VERIFY RCS pressure to between MNPSH and 265 psia (C–03, PPC).

___/___ 4.4.15 Using "SDC SYS HX FLOW CNTL, HIC–3657" (C–01) and "SDC SYS TOTAL FLOW, FIC–306", ADJUST SDC System ΔT to obtain value calculated in step 4.4.7 for concurrent SDC/RCP operation with 0°F heatup rate.

___/___ 4.4.16 As necessary, STABILIZE RCS temperature OR CONTINUE plant heatup.

___/___ 4.4.17 CLOSE manual disconnect switch, 89–SI652 (for SI–652, "SDC SYS SUCT CTMT ISOL," west wall of Control Room).

___/___ 4.4.18 CLOSE "2–SI–651, MANUAL DISCONNECT SWITCH, NSI651," (14′6″ Aux Bldg, west wall, across from B51 enclosure)

Attachment 6
Plant Heatup Conditional Actions
(Sheet 1 of 5)

1. IF in MODE 3 AND one RCP loop is *not* operable, LOG in to Tech Spec Action statement 3.4.1.2.
2. IF at any time, RCP operation *cannot* continue, and it is necessary to restore SDC, PERFORM the following:
   - 2.1 PERFORM one of the following:
     - STOP *affected* RCPs.
     - IF less than two RCPs will remain running, STOP *all* RCPs and LOG into the following.
       - MODE 3, TSAS 3.4.1.2 ACTION b
       - MODE 4, TSAS 3.4.1.3 ACTION c (10)
     - Refer To applicable RCP NPSH Attachment, or PPC NPSH display, and CHECK NPSH for running RCPs is met.
       - IF NPSH for running RCPs is *not* met, STOP RCPs.
   - 2.2 IF applicable, Refer To *one* of the following and COOLDOWN plant to less than 300°F.
     - EOP 2528, "Loss of Offsite Power/Loss of Forced Circulation" (2)
     - OP 2207, "Plant Cooldown"
   - 2.3 VERIFY all HPSI pump control switches are in "PULL TO LOCK."
3. IF *at any time*, during heatup, one RCP is lost (initially two RCPs operating AND SDC *not* in service), PERFORM the following:
   - 3.1 TRIP remaining RCP.
   - 3.2 LOG into the following: (10)
     - MODE 3, TSAS 3.4.1.2 ACTION b
     - MODE 4, TSAS 3.4.1.3 ACTION c
   - 3.3 Refer To EOP 2528, "Loss of Offsite Power/Loss of Forced Circulation."
   - 3.4 ADJUST RCS pressure to establish adequate NPSH for RCP operation as specified in Attachment 4.
   - 3.5 IF 2 RCPs are available to be operated, Refer To OP 2301C and START 2 RCPs.
     - 3.5.1 LOG out of the following:
       - MODE 3, TSAS 3.4.1.2 ACTION b
       - MODE 4, TSAS 3.4.1.3 ACTION c

Question #: **12** | **Question ID: 9600016** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☑ Lower Order?

I-SRO Ques. # 12 | Rev. 0 | ☑ Selected for Exam | **Origin: Mod** | ☐ Past NRC Exam?

A Rapid Downpower at 50%/hr is in progress due to an RCS leak in containment that exceeds the administrative limit. The following plant conditions presently exist:

* Plant power is 93% and dropping at the intended rate.
* Pressurizer level is 65% and stable.
* RCS pressure is 2250 psia and stable.
* One charging pump is running, Letdown is at approximately 30 gpm.
* Adding boric acid to the charging pump suction to maintain the desired rate of power reduction.
* Forcing Pressurizer Sprays in progress.
* C02/3 annunciator in alarm; D-37, "PZR PRESSURE SELECTED CHANNEL DEVIATION HI/LO"
* Channel "Y" Pressurizer Level and Pressure controlling normally.

Then, during the load reduction, Pressurizer Level Channel "X" fails to zero (0) and the following occur:

* All control systems respond as designed to the failure.
* C02/3 annunciator in alarm; A-38, "PRESSURIZER CH X LEVEL HI/LO".
* C02/3 annunciator in alarm; C-38, "PRESSURIZER CH X LEVEL LO-LO".
* PPC alarms on Monitor #2 indicative of the instrument failure.

Which of the following actions must the Unit Supervisor direct <u>and</u> why?

☑ **A** Per the ARP for C-38; shift pressurizer heater control to channel "Y" and restore pressurizer heaters, to ensure adequate margin from DNB is maintained.

☐ **B** Per SP-2602A RCS Leakage; deselect Pressurizer Level Channel "X" from the leak rate calculation, to ensure valid trending of RCS Leak Rate by the PPC.

☐ **C** Per the ARP for A-38; place the standby charging pumps in "Pull-To-Lock", to prevent the rate of the plant downpower from accelerating above 50%/hr.

☐ **D** Per AOP-2575, Rapid Downpower; shift pressurizer heater control to channel "Y" and restore the pressure controller setpoint, to prevent a plant trip on TM/LP.

Justification

A - CORRECT: Even though Ch. "Y" is the controlling channel of PZR pressure, Ch. "X" failing to zero will trip <u>all</u> PZR heaters. The heater control switch must be selected to ignore the failed channel and the heater breakers must be <u>manually</u> reclosed (they will <u>not</u> auto close even though the automatic controls are calling for more heater output). If this is action is not taken in a very timely manner, RCS pressure will drop below the minimum required by Tech. Spec. to ensure adequate DNB margin, due to the lower pressure control setpoint necessary to "Forcing PZR Sprays".

B - WRONG: The high rate of power change exceeds the PPC program capabilities for calculating an accurate RCS leak rate. <u>Plausible</u>: The failed PZR instrument could impact the RCS leak rate calculation and potentially be of concern if the rate of power change were less.

C - WRONG: The failed instrument effects heaters only and has no effect on the standby pumps. <u>Plausible</u>: The alarms received for this failure would be identical to those received had the instrument been aligned to automatically start the standby charging pumps, as implied by the ARP. If this occurred, the rate of power drop would accelerate dramatically.

D - WRONG: <u>All</u> PZR heaters are effected by this failure, not just the Backup heaters (as in a loss of control power). Restoring the setpoint to a normal setting will not recover RCS pressure and the plant will trip on TM/LP. <u>Plausible</u>: Although the AOP gives guidance to force sprays, it does not allow for heater recovery on a failed channel. OP-2204, Load Changes contains the detailed guidance used by operators globally to commence, and secure from, forcing pressurizer sprays. However, this guidance simply states to adjust controller setpoint, as necessary, to maintain pressure at the desired value.

References

ARP-2590B-2150 Alarm C-38, PZR Level Lo-Lo

Comments and Question Modification History

Bob K. - D-3/C (change to match given reactivity plan)
Modified stem and choice "C" to reflect only one charging pump selected to run, others in standby Also changed the downpower rate to 50%/hr to match the applicable Reactivity plan. - RLC
Bill M. - D-3/C, K (Able to rule out distaractor C due to downpower rate of 30%/hr, which does not match the stem. Change to 50%/hr. Distractor more plausible. May have resulted in a 50/50)
Changed downpower rate in distractor "C" to 50%/hr per recommendation. RJA
Angelo - D-3/C; No comments.

NRC K/A System/E/A **System** 004 Chemical and Volume Control System

Question #: **12** | **Question ID: 9600016** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

I-SRO Ques. # **12** | Rev. 0 | ☑ Selected for Exam | **Origin: Mod** | ☐ Past NRC Exam?

Number A2.15 **RO** 3.5 **SRO** 3.7 **CFR Link** (CFR: 41.5/ 43/5 / 45/3 / 45/5)

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

Approval Date 02/16/04 Effective Date 03/04/04

Setpoint: 20%

C-38

PRESSURIZER CH X LEVEL LO LO

AUTOMATIC FUNCTIONS

1. IF "SEL SW" is in "X+Y" position, *all* heaters de-energize.

CORRECTIVE ACTIONS

1. OBSERVE actual level on pressurizer level recorder, LR-110, pressurizer level controllers (C-03) and PPC.
2. IF annunciator is *not* valid, SHIFT pressurizer level control to channel "Y."
   2.1 SHIFT pressurizer heater control "SEL SW" to channel Y.
3. VERIFY the following:
   - Available backup charging pumps are running (C-02).
   - Letdown flow is at minimum of 28 gpm on "LTDN FLOW, FI-202" (C-02).
4. IF level *cannot* be restored or continues to lower, Refer To AOP 2568, "Reactor Coolant System Leak."
5. WHEN annunciator clears, VERIFY *all* required heaters energize.
6. WHEN level rises to 4% below setpoint, VERIFY second back up charging pump stops.
7. WHEN level rises to 3% below setpoint, VERIFY first back up charging pump stops.
8. VERIFY level is restored to setpoint.
9. IF alarm was caused by channel X malfunctioning, SUBMIT Trouble Report to I&C Department.
10. Refer To Technical Specifications LCOs 3.3.3.5 and 3.3.3.8 to determine ACTION Statement requirements.

SUPPORTING INFORMATION

1. Initiating Devices
   - LC-110XL
2. Computer Points
   - L110X
3. Possible Causes
   - Controller malfunction
   - RCS inventory loss
4. Technical Specifications LCOs: 3.4.4, 3.3.3.5 and 3.3.3.8
5. Procedures
   - OP 2304A, "Volume Control Portion of CVCS"
   - AOP 2568, "Reactor Coolant System Leak"
6. Control Room Drawings
   - 25203-32007, sh. 57
7. Annunciator Card Location: TB10-J12

ARP 2590B-215
Rev. 000
Page 1 of 1

A-38

8. IF *indicated* high or low level was caused by controller or transmitter malfunction (other than Reactor Regulating System inputs), PERFORM the following:
   - 8.1 SHIFT "LTDN FLOW CNTL, HIC-110" in "MAN" (C-02).
   - 8.2 ADJUST "LTDN CNTL, HIC-110" to stabilize Pressurizer level and Letdown flow (C-02).
   - 8.3 IF desired, COMMENCE forcing Pressurizer sprays.
   - 8.4 SHIFT Pressurizer level control to channel "Y" (C-03).
   - 8.5 RESTORE Letdown to automatic as follows:
     - 8.5.1 ADJUST bias to "0", using black thumbwheel.
     - 8.5.2 SHIFT "LTDN FLOW CNTL, HIC-110" to "AUTO."
     - 8.5.3 ADJUST bias to restore Pressurizer level to setpoint.
     - 8.5.4 SHIFT Pressurizer heater control "SEL SW" to channel "Y."

9. As necessary, RESET the following Pressurizer heater breakers:
   - "PROP HTR GROUP 1"
   - "PROP HTR GROUP 2"
   - "BACKUP HTRS GROUP 1"
   - "BACKUP HTRS GROUP 2"
   - "BACKUP HTRS GROUP 3"
   - "BACKUP HTRS GROUP 4"

10. IF instrument malfunction is determined *not* to be the cause of low level, Refer To the following, as applicable:
    - AOP 2512, "Loss of All Charging"
    - AOP 2568, "Reactor Coolant System Leak"
    - AOP 2569, "Steam Generator Tube Leak"

11. IF actual level was high or low, VERIFY level is restored to normal.

12. IF alarm was caused by channel "X" malfunctioning, SUBMIT Trouble Report to Instrumentation & Control Department.

ARP 2590B-213
Rev. 001-00
Page 3 of 4

Question #: **13** | **Question ID:** **53730** | ☐ RO ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

I-SRO Ques. # 13 | Rev. 4 | ☑ Selected for Exam | **Origin:** **Bank** | ☐ Past NRC Exam?

A reactor startup is in progress using CEA withdrawal. The RO has just stopped withdrawing Group # 7 CEAs and makes the following announcements:

* The reactor is critical.
* Startup rate is positive and stabilizing at ~1.5 DPM.

Which of the following actions should the Reactivity SRO direct?

☐ **A** Commence Emergency Boration until the reactor is subcritical.

☐ **B** Insert the Group #7 CEAs to lower the startup rate below 0.5 DPM.

☑ **C** Trip the reactor and Go to EOP 2525, "Standard Post Trip Actions".

☐ **D** Insert all CEAs per OP-2206, "Reactor Shutdown" and notify RE.

Justification

C - CORRECT OP 2202, "Reactor Startup " Conditional Actions require <u>tripping</u> the reactor, and transitioning to EOP-2525, if a SUR of 1.0 DPM is sustained.

A - WRONG: This action may be acceptable if an abnormal count rate was due to an uncontrolled cooldown during the reactor startup.
<u>Plausible</u>: This action would stop the power rise and shutdown the reactor, but it is unacceptable with a high SUR.

B - WRONG: This is acceptable if SUR has not yet exceeded 1.0 DPM.
<u>Plausible</u>: Correct action if SUR <u>briefly</u> spiked above 1.0 DPM or, stabilized just below 1.0 DPM.

D - WRONG: To slow for a high startup rate, even if it is just barely above the threshold for "excessive".
<u>Plausible</u>: Correct action if criticality were occurring earlier than predicted (outside of acceptable limits).

References

OP-2202; Pg. 36, Attach. 5, Rx Startup Conditional Actions

Comments and Question Modification History

Bob K. - D-3/W (Did not remember "trip" criteria)
Bill M. - D-2/W, K (Did not remember the actual trip criteria value.)
Angelo - D-3/C; No comments.

NRC K/A System/E/A **System** 012 Reactor Protection System

Generic K/A Selected

NRC K/A Generic **System** 2.4 Emergency Procedures /Plan

Number 2.4.4 **RO** 4.5 **SRO** 4.7 **CFR Link** (CFR: 41.10 / 43.2 / 45.6)

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Attachment 5
Reactor Startup Conditional Actions
(Sheet 1 of 3)

1. IF, at any time, the following conditions occur, PERFORM the specified action:
   - IF T_{avg} lowers to between 515 and 525 °F AND the reactor is critical, Refer to OP 2619A–001, "Control Room Daily Surveillance," and RECORD RCS temperature once every hour.
   - IF T_{avg} lowers to less than 515 °F AND the reactor is critical, PERFORM the following:
     - RAISE T_{avg} to greater than 515 °F within 15 minutes.
     - IF T_{avg} is *not* greater than 515 °F within 15 minutes, PLACE plant in HOT STANDBY condition within the next 15 minutes.
     - Refer To T/S LCO 3.1.1.5 and DETERMINE applicability.
   - IF an uncontrolled cooldown occurs (T_C less than 500 °F), PERFORM the following:
     - TRIP reactor and INITIATE EOP 2525, "Standard Post Trip Actions."
     - STOP one of the 4 operating RCPs (C–04).
     - Refer To AOP 2558, "Emergency Boration," and INITIATE emergency boration.
     - Refer To T/S LCO 3.4.9.1 and DETERMINE applicability.
2. IF at any time a sustained SUR of 1.0 dpm is achieved, **TRIP** reactor and Go To EOP 2525, "Standard Post Trip Actions."
3. IF at any time during reactor startup, it appears that criticality is reached, or is predicted to be reached, outside plus or minus 0.5% $\Delta\rho$ (0.9% $\Delta\rho$ for initial startup after refueling) band of ECP, PERFORM the following:

   3.1 INSERT *all* CEA *regulating* groups in sequence (C–04).

   3.2 REQUEST Chemistry Department sample and determine RCS boron concentration.

   3.3 INITIATE a CR for Reactivity Management tracking.

   3.4 Refer To OP 2208, "Reactivity Calculations" and, independent of CEA position, VERIFY adequate SHUTDOWN MARGIN using OP 2208–013, "Shutdown Margin Determination."

   3.5 NOTIFY Reactor Engineering.

Question #: **14** | **Question ID: 9000021** | ☐ **RO** | ☑ **SRO** | ☐ Student Handout? | ☑ Lower Order?

I-SRO Ques. # 14 | Rev. 0 | ☑ Selected for Exam | **Origin:** **New** | ☐ Past NRC Exam?

The plant is in Mode 6 with the following conditions:

* Core re-load in progress and approximately half way completed.
* "A" LPSI pump running for Shutdown Cooling (SDC) operation.
* "B" LPSI pump in standby, aligned for SDC use.
* "A" train of Spent Fuel Pool (SFP) cooling in service.

Then, "A" LPSI pump is lost due to a breaker fault. When the "B" LPSI pump is started, it seizes and trips on breaker overload.
The Unit Supervisor (US) then directs the RO to recover SDC using the "B" Containment Spray (CS) pump.

Which of the following additional directions must the US give, while SDC flow is being supplied by a CS pump?

☑ **A** All fuel movement in containment must remain secured.

☐ **B** SDC supplementing of SFP cooling must be secured.

☐ **C** Containment must remain evacuated of non-essential personnel.

☐ **D** Containment Closure must be fully set with all access doors closed.

Justification

A - CORRECT; When SDC is being supplied by a CS pump, does not constitute an OPERABLE train of SDC. Therefore, all fuel movement in CTMT must be secured.

B - WRONG; A CS pump has the capacity to supply SDC flow and supplement SFP cooling, but just barely, if a train of SFP cooling is in operation.
Plausible; Examinee may believe with the limited capacity of a CS pump (compared to a LPSI pump), supplementing SFP cooling is not possible (it would not be at the beginning of the outage).

C - WRONG; CTMT evacuation would probably occur if a CS pump needed to be used for recovery as is required if SDC flow cannot be restored in 15 minutes. However, once the CS pump restores flow, evacuation is no longer necessary.
Plausible; Examinee may realize SDC is not considered fully operational being supplied by a CS pump and, therefore, require evacuation of CTMT be maintained. This is true if a plant heatup to over 190°F occurs.

D - WRONG; CTMT Closure must be set on initial loss of SDC flow, but once heat removal is regained, it may stop.
Plausible; Examinee may believe that with a CS pump supplying RHR needs (SDC not operable), CTMT closure must be maintained.

References

AOP-2572, Pg. 3, Discussion Section and Pages 28 & 31

Comments and Question Modification History

Bob K. - D-3/W (Did not remember SDC not Operable with CS Pump supplying).
Corrected a typo in Distractor B. - RJA
Bill M. - D-4/W, G (Did not realize that SDC was inoperable, which requires CORE ALTERATIONS to remain suspended.)
Angelo - D-5/C. Difficult but fair.

NRC K/A System/E/A System 026 Containment Spray System (CSS)

Generic K/A Selected

NRC K/A Generic System 2.4 Emergency Procedures /Plan

Number 2.4.9 **RO** 3.8 **SRO** 4.2 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

1.0 PURPOSE

1.1 Objective

This procedure provides actions for recovering from a partial or total loss of shutdown cooling.

1.2 Discussion

During SDC operation, there may *not* be flow past the loop RTDs. Core inlet and outlet temperatures are accurately measured during those conditions using SDC to RCS temperature, T351Y, and RCS to SDC temperature, T351X, respectively. The average of these indicators provides a temperature that is equivalent to the average RCS temperature in the core.

Containment Closure is established when all of the following conditions exist:

- The equipment door is closed and held in place by a minimum of four bolts.
- A minimum of one door in each airlock is closed.
- Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:
  - Closed by a manual or automatic isolation valve, blind flange, or equivalent, or
  - Capable of being closed under administrative control

The use of the CS pump for decay heat removal does not meet the definition of an Operable SDC train (LCO 3.9.8). Therefore no fuel movement is permitted when a CS pump is aligned to SDC per this procedure.

1.3 Applicability

This procedure is applicable in MODEs 4, 5, 6 and Defueled.
Use of the CS pumps is limited to MODE 6 and Defueled.

INSTRUCTIONS

5.10 (continued)

2) CHECK pressure at 2-RW-66, "SFPC/RW Purification Return Sample Stop," less than 30 psig.

CONTINGENCY ACTIONS

2) IF pressure at 2-RW-66 is greater than 30 psig, PERFORM the following:

a. THROTTLE 2-RW-15, "SDC to SFPC Stop" to obtain less than 30 psig at 2-RW-66.

b. CONTACT Engineering for additional guidance on decay heat removal.

Table 1.0

Fuel Assemblies Off-Loaded	Flow splits with SFPC in service		Flow splits with SFPC *not* in service	
	Flow to RFP	Flow to SFP	Flow to RFP	Flow to SFP
0-80	1700	0	1250	450
81-170	1400	300	750	950
171-217	1100	600	400	1300

b. IF a CS pump is in service, PERFORM the following:

1) THROTTLE the following valves to obtain Table 1.0 flow splits:

- SI-615, "LPSI INJ VLVS" LOOP 1A
- SI-625, "LPSI INJ VLVS" LOOP 1B
- SI-635, "LPSI INJ VLVS" LOOP 2A
- SI-645, "LPSI INJ VLVS" LOOP 2B
- 2-RW-15, "SDC to SFPC Stop"

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

2–SI–306 has designed leakby that diverts flow around the SDC heat exchangers and could challenge heat removal with CS pumps supplying SDC.

___5.17 IF a CS pump is in service on SDC and sufficient cooling cannot be obtained with 2–SI–306 closed, CLOSE the applicable LPSI to SDC heat exchanger isolation valve:

- 2–SI–452, LPSI Pump Discharge to "A" SDC Heat Exchanger
- 2–SI–453, LPSI Pump Discharge to "B" SDC Heat Exchanger

___5.18 REPEAT steps 5.12 through 5.17 as needed to control RCS temperature.

___5.19 WHEN ready to shift SDC from a CS pump to a LPSI pump, PERFORM Attachment 8, "Realigning LPSI to Supply SDC and SFPC."

___5.20 WHEN RCS pressure is stable AND RCS temperature is less than 200°F and stable, STOP Containment Closure activities.

___5.21 Go To Section 10.0.

Question #: **15** | **Question ID: 9000010** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

1-SRO Ques. # **15** | Rev. **0** | ☑ Selected for Exam | **Origin: New** | ☐ Past NRC Exam?

The required surveillance must be performed after repairs were made to the "A" Service Water Strainer Flush Valve, 2-SW-90A. The completed surveillance indicates the valve stroke time is slightly above the Maximum 'Normal Limit", but is below the Maximum "Acceptable Limit". All other parameters are within the "Acceptable Limits".

Which of the following describes the condition of the "A" Service Water Pump <u>and</u> the required action?

☐ **A** The "A" Service Water Pump remains inoperable. Perform the required repairs to the "A" Service Water Strainer Flush Valve, 2-SW-90A, then perform the required surveillances to restore the "A" Service Water Pump to OPERABLE.

☐ **B** The "A" Service Water Pump remains inoperable. Obtain a different set of test equipment and immediately retest the "A" Service Water Strainer Flush Valve, 2-SW-90A, again to verify that the previous data was accurate.

☐ **C** The "A" Service Water Pump is considered OPERABLE. Place the "A" Service Water Pump Strainer in an Augmented Testing Program and test weekly to ensure the "A" Service Water Strainer Flush Valve, 2-SW-90A, remains OPERABLE.

☑ **D** The "A" Service Water Pump is considered OPERABLE. If the "A" Service Water Strainer Flush Valve, 2-SW-90A, exceeds the Maximum "Normal Limit" on an immediate retest, then declare the "A" Service Water Pump inoperable.

Justification

D IS CORRECT; Per SP 2612A, "A" Service Water Pump Tests, Attachment 2, the first failure of a valve stroke test to be within the "Normal" limits does not render that component inoperable; however, an immediate retest must be performed. If a second failure of "Normal" stroke time limit has occurred, then the component is inoperable.

A is incorrect; Even though the Service Water Pump did not meet the "Normal" limit criteria it may be considered OPERABLE; however, a second set of data must be taken. Repairs are NOT required.
<u>Plausible</u> because the examinee may consider the component inoperable from the first set of failed data. If the component is considered inoperable, then typically, a component must be repaired to restore it to OPERABLE.

B is incorrect; Even though the Service Water Pump did not meet the "Normal" limit criteria the surveillance procedure allows it to be considered OPERABLE. Different test equipment may be obtained to verify the previous data.
<u>Plausible</u> because the examinee may consider the component inoperable from the first set of failed data. Verifying previous data with different test equipment is allowed by procedure and is correct.

C is incorrect; The surveillance procedure allows the Service Water Pump to be considered OPERABLE. The Service Water Pump may be tested on a more frequent basis; however, the stoke time must be performed immediately after the initial failure to meet the "Normal" limit.
<u>Plausible</u> because the examinee may not be aware of the need to immediately perform a second stoke time test.

References

SP 2612A, "A" Service Water Pump Tests

Comments and Question Modification History

Bob K. - D-3/W (Reword answer to say "immediate retest").
Reworded correct answer to include the term "immediate retest". RJA
Bill M. - D-3/W, K (Didn't realize an immediate retest was required.)
Angelo - D-4/C; No comments.

NRC K/A System/E/A **System** 076 Service Water System (SWS)

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.12 **RO** 3.7 **SRO** 4.1 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of surveillance procedures.

Question #: 15 | Question ID: 9000010 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

1-SRO Ques. # 15 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

4.1.5 IF testing 2–SW–90A, "A" SERVICE WATER PUMP STRAINER FLUSH," PERFORM the following:

NOTE

Two Operators are required to time strainer flush valves due to location of switches and quick operation of valves.

_______ a. Refer To OP 2328A, "Sodium Hypochlorite System," and VERIFY Sodium Hypochlorite Injection to "A" Service Water Pump is terminated. | 5

_______ b. LOG ENTRY into TSAS 3.7.4.1 and TRMAS 7.1.21.A. | 3 5

_______ c. IF, at any time, valve does *not* stroke fully, Go To Attachment 1.

_______ d. PLACE "A" Service Water Pump Strainer control switch in "HAND" (C–58A). | 4

_______ e. PRESS "A" Service Water Pump Strainer "START" button and MEASURE open stroke time.

_______ f. RECORD 2–SW–90A open stroke time on SP 2612A–003.

_______ g. PRESS "A" Service Water Pump Strainer "STOP" button and MEASURE close stroke time.

_______ h. RECORD 2–SW–90A close stroke time on SP 2612A–003.

_______ i. PLACE 'A' Service Water Pump Strainer control switch in "AUTO" (C–58A). | 4 | 3

_______ j. DOCUMENT 2–SW–90A "Normal" limits Results on SP 2612A–003.

_______ k. DOCUMENT 2–SW–90A Operational Readiness Results on SP 2612A–003.

_______ l. IF Operational Readiness Results "UNSAT," Go To Attachment 1.

_______ m. IF "Normal" limits Results "UNSAT," Go To Attachment 2.

_______ n. LOG EXIT from TSAS 3.7.4.1 and TRMAS 7.1.21.A. | 5

Attachment 2
Actions for IST Data Outside "Normal" Limits

(Sheet 1 of 1)

NOTE

The first failure of a valve stroke time test to be within IST "Normal" limits does *not* render that component INOPERABLE, but an immediate retest must be performed. 7 3

1. IF a second failure of "Normal" stroke time limit has occurred, Go To Attachment 1. 7 3
2. VERIFY the following meet test requirements:
   - Test prerequisites
   - System conditions
   - Procedure performance
3. REVIEW recorded data and DETERMINE if test equipment is providing accurate information.
4. To retest component, PERFORM the following:

   4.1 OBTAIN *new* applicable Form data sheets (new cover sheet *not* required).

   4.2 ENTER the following on applicable OPS Form cover sheet "Comments" section:

   "Retest of (specify component) required, additional data sheets attached."

   4.3 INDICATE on *new* OPS Form data sheets that data is from retest and ATTACH to original Form.

   4.4 Go To applicable section of this procedure and PERFORM retest.

Attachment 1
Actions for IST Data Outside "Acceptable" Limits
(Sheet 1 of 1)

1. CONSIDER component *not* OPERABLE and NOTIFY SM or US. 3
2. IF in MODE 1, 2, 3 or 4, LOG into TS 3.7.4.1, and TRMAS 7.1.21 A as required. 5
3. SUBMIT CR and RECORD CR number in applicable Form cover sheet.
4. NOTIFY the following:
   - IST Coordinator
   - System Engineer

Question #: 16 | Question ID: 9000012 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
I-SRO Ques. # 16 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

The plant is in MODE 6 with the following conditions:

- Fuel movement is in progress.
- The Personnel Airlock Doors are open
- The Equipment Hatch is open.
- Containment Purge is in operation.
- Containment Atmosphere Radiation Monitor, RM-8123, is out of service for repairs.

The Auxiliary Building PEO has just reported that the blower for Containment Atmosphere Radiation Monitor, RM-8262, has tripped and is very hot to the touch.

Which of the following actions must be taken and why?

☐ **A** Immediately suspend CORE ALTERATIONS and establish Containment Closure prior to resuming fuel movement, to ensure a potential fuel handling accident in Containment is NOT released to the environment.

☐ **B** Immediately suspend CORE ALTERATIONS and restore the Radiation Monitor blower prior to resuming fuel movement, to ensure a potential fuel handling accident in Containment is NOT released to the environment.

☑ **C** Ensure a control room operator is specifically assigned to close the Containment Purge Valves within 30 minutes of an event, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.

☐ **D** Restore the Containment Purge Valves to OPERABLE status within the next 30 minutes or immediately close the Purge Valves, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.

Justification

C IS CORRECT; TS 3.9.4 requires that Containment Purge Valves either be closed by an automatic isolation or be capable of being closed under administrative control. This means that a specific individual is designated as available to close the Purge Valves within 30 minutes of a fuel handling accident in Containment.

A is incorrect; CORE ALTERATIONS do NOT need to be suspended and Containment Closure is still available.
Plausible if the examinee believes that the Purge Valves need to be closed by an automatic isolation signal. (Only one Containment Radiation Monitor needs to be OPERABLE to initiate and automatic closure of all 4 Purge valves.) The examinee may also believe that the loss of the only remaining Radiation Monitor (and automatic isolation of the Purge Valves) results in a loss of Containment Closure. (Containment Closure must be set or available during CORE ALTERATIONS.)

B is incorrect; CORE ALTERATIONS do NOT need to be suspended; however, it would be appropriate to have the Radiation Monitor blower repaired.
Plausible if the examinee believes that the Purge Valves need to be closed by an automatic isolation signal.

D is incorrect. In MODE 6, the Purge Valves are still considered OPERABLE even if they are NOT able to be closed by an automatic isolation signal.
Plausible because Tech Spec 3.6.3.1 requires each Containment Isolation Valve to be OPERABLE (in MODES 1, 2, 3, and 4). These valves are demonstrated OPERABLE by verifying the automatic signal functions or the valves are closed and secured. This Spec does NOT apply to the Containment Purge Valves in MODE 6.

References

Tech. Spec. 3.9.4 LCO; Containment Penetrations

Comments and Question Modification History

Bob K. - D-3/C (Change "Designate" to "Ensure" for control room operator).
Minor rewording of choices "C" and "D" per above comments - RLC
Bill M. - NOT VALIDATED. Inadvertently selected the answer for #17 and did not see this question. When discussed afterwards, Bill felt that this was an LOD of 3 and that he would have known the correct answer.
Angelo - D-3/C; No comments.

NRC K/A System/E/A System 029 Containment Purge System (CPS)

Generic K/A Selected

NRC K/A Generic System 2.1 Conduct of Operations

Question #: **16** | **Question ID: 9000012** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
I-SRO Ques. # 16 | Rev. 0 | ☑ Selected for Exam | **Origin: New** | ☐ Past NRC Exam?

Number 2.1.32 **RO** 3.8 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.2 / 45.12)
Ability to explain and apply system limits and precautions.

September 20, 2004

REFUELING OPERATIONS

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door shall be either:

1. closed and held in place by a minimum of four bolts, or

2. open under administrative control* and capable of being closed and held in place by a minimum of four bolts,

b. The personnel air lock shall be either:

1. closed by one personnel air lock door, or

2. capable of being closed by an OPERABLE personnel air lock door, under administrative control *, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. Be capable of being closed under administrative control *

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel assemblies in the containment.

* Administrative controls shall ensure that appropriate personnel are aware that the equipment door, personnel air lock door and/or other containment penetrations are open, and that a specific individual(s) is designated and available to close the equipment door, personnel air lock door and/or other containment penetrations within 30 minutes if a fuel handling accident occurs. Any obstructions (e.g., cables and hoses) that could prevent closure of the equipment door, a personnel air lock door and/or other containment penetration must be capable of being quickly removed.

MILLSTONE - UNIT 2 3/4 9-4 Amendment No. ~~60~~, ~~85~~, ~~98~~, ~~245~~, 284

Question #: 17 | **Question ID: 9079010** ☐ RO ☑ SRO ☐ Student Handout? ☐ Lower Order?
I-SRO Ques. # 17 | Rev. 0 ☑ Selected for Exam **Origin: Mod** ☐ Past NRC Exam?

The plant was at 100% power when CONVEX ordered Main Generator output be lowered from 900 MWe to 600 MWe in 15 minutes.
AOP 2557, "Emergency Generation Reduction", was initiated and the following conditions now exist:

* Group 7 CEAs are at 170 steps withdrawn.
* Main Generator output is 610 MWe and slowly lowering.
* "A" Steam Dump Bypass Valve is 75% open and stable.
* "BYPASS TO CND", PIC-4216 output is 83% and stable.
* "B", "C" and "D" Steam Dump Bypass Valves are open 75% and stable.
* "STEAM DUMP TAVG CNTL", TIC-4165 output is 85% and stable.
* "RC LOOP 1 COLD LEG TEMP HI" annunciator has just alarmed (C02/3; C-34).
* "RC LOOP 2 COLD LEG TEMP HI" annunciator has just alarmed (C02/3; D-34).
* RCS Tcold is 550 °F and slowly rising (RPS).

Which one of the following actions should the US direct?

☐ **A** Transfer control of the steam dumps to Foxboro IA control and lower Tcold to program.

☑ **B** Lower the setpoint on the "A" steam dump and log into the DNB Technical Specification.

☐ **C** Immediately trip the reactor and go to EOP 2525, "Standard Post Trip Actions".

☐ **D** Insert CEAs until Tcold is back on program and both C02/3 alarms have cleared.

Justification

B - CORRECT; Turbine load has been lowered ahead of the "A" Steam Dump Valve controller setpoint, as indicated by TIC-4165 output being higher than PIC-4216. Therefore, raising the output of PIC-4216 to open the "A" steam dump is the most immediate action to restore Temperature to program. This is an expected possible action if the setpoint on PIC-4216 is not lowered enough initially. However, Tcold is already above the DNB Tech. Spec. limit, so the LCO must be entered.

A - WRONG; Transferring the "A" steam dump to Foxboro IA control would immediately fail the valve closed, making things far worse. Plausible: This action is prudent if upon initiation of the procedure, it was noted the controller on C05 was not responding properly. The indications given show the C05 controller may not be operating correctly, when in fact, the setpoint must be lowered to ensure the "A" steam dump stays ahead of the other 3 valves.

C - WRONG; AOP 2557 requires RCS temperature to be maintained within 10°F of program or a plant trip is required. AOP 2557 maintains reactor power constant, therefore, Tcold should be ~545°F per Attachment 1. RPS indication (and C02/3 alarms) indicate Tcold is >/=549°F, which is < 10°F above program value.
Plausible; If plant power level is extracted from the Main Generator output, then Tcold should be ~540°F. This would mean that Tcold is > 10°F above the program value and a trip is required.

D - WRONG; Driving in CEAs will lower RCS temperature by lowering reactor power but, RCS temperature is "out-of-program" because generator load reduction was not controlled properly.
Plausible: This is an acceptable action if temperature is out of band due to turbine load reduction being ahead of reactor power reduction. However, reactor power is not being reduced, by procedure.

References

AOP 2537, Emergency Generation Reduction, Pages 8 & 9

Comments and Question Modification History

Bob K. - D-3/W (Did not read "2557" and thought Rx power was being reduced. Upon second thought, was able to dertermine the correct answer.)
Bill M. - D-3/W, K (Doesn't feel that picking up load on the Turbine is appropriate for this condition. Feels that adjusting Condenser Dumps is more appropriate.
Revised distractor "A" slightly to transfer control of all the steam dumps to the Foxboro IA vs only the "A" steam dump. Revised "B" to lower the setpoint on the "A" steam dump vs pick up load on the Turbine. - RJA
Mike C. - /W, Lower controller outputs in stem to put valves at 75% open. **Done - RLC**
Angelo - D-4/C; No comments.

NRC K/A System/E/A System 045 Main Turbine Generator (MT/G) System

Generic K/A Selected

NRC K/A Generic System 2.4 Emergency Procedures /Plan

Question #: 17 | Question ID: 9079010 | ☐ RO ☑ SRO | ☐ Student Handout? | ☐ Lower Order?
I-SRO Ques. # 17 | Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

Number 2.4.47 **RO** 4.2 **SRO** 4.2 **CFR Link** (CFR: 41.10,43.5 / 45.12)

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Millstone Unit 2 Emergency Generation Reduction	AOP 2557 Revision 006 – 06 Page 8 of 16

INSTRUCTIONS | CONTINGENCY ACTIONS

CAUTION

1. PPC calorimetric is inaccurate due to SG level transients. The most accurate available indication of reactor power is ΔT power.
2. Turbine exhaust hood temperature greater than or equal to 225°F requires manual turbine trip. ②

___3.9 WHEN transferring steam load from main turbine to steam dump and bypass valves, MONITOR the following:

- ΔT power
- Condenser backpressure (PPC point P5127)
- Turbine exhaust hood temperature (UR 4500 "TURBINE TEMP & EXPANSION," recorder points 8 and 9, PPC points T4319 and T4320)
- Condensate header flow and pressure
- MVARs

___3.10 WHEN reducing turbine load, MAINTAIN "A" steam dump bypass valve 20 to 100% open as follows:

- Using "STM DUMP TAVG CNTL, TIC-4165", THROTTLE open "B," "C," and "D" steam dump bypass valves

Level of Use Continuous

STOP THINK ACT REVIEW

Millstone Unit 2 Emergency Generation Reduction	AOP 2557 Revision 006–06 Page 9 of 16

INSTRUCTIONS	CONTINGENCY ACTIONS
___3.11 Using one of the following, ADJUST turbine to desired load: • "LOAD LIMIT POT" • "LOAD SELECTOR, INCREASE" and "LOAD SELECTOR, DECREASE" buttons	

NOTE

Receipt of annunciators DA-37, "HI COND D/T" AND DB-37, "HI COND DIS TEMP" is expected during this evolution (C-06/07).

INSTRUCTIONS	CONTINGENCY ACTIONS
___3.12 IF annunciator DA-37 (C-06/07), "HI COND D/T" or DB-37 (C-06/07), "HI COND DIS TEMP," is received, Refer To ARP 2590E, "Alarm Response for Control Room Panels, C-06/7."	
___3.13 WHEN desired load is achieved, STABILIZE turbine load and RCS temperature.	
___3.14 ENSURE pressurizer level 35 to 70%.	3.14.1 IF the pressurizer level control system is *not* operating properly in automatically, RESTORE and MAINTAIN pressurizer level 35 to 70% by performing ANY of the following: a. OPERATE the pressurizer level control system. b. Manually OPERATE charging and letdown.

Level of Use
Continuous

STOP THINK ACT REVIEW

Question #: **18**	***Question ID:* 9000011**	☐ **RO**	☑ **SRO**	☐ **Student Handout?**	☐ **Lower Order?**
I-SRO Ques. # **18**	**Rev. 0**	☑ **Selected for Exam**	***Origin:* New**		☐ **Past NRC Exam?**

The plant is in normal operation at 100% power, when a Fire System Trouble annunciator is received on C06/7 and Zone 45 on Fire Panel, C-26. The Auxiliary Building PEO subsequently calls from the West DC Switchgear Room and reports the following:

* One Ion Chamber smoke detector is in alarm.
* The Halon strobe lights and horn are pulsating slowly.
* All other smoke detectors are operating normally (not in alarm).
* There is no smoke or fire in the area. The detector appears to have failed.

Which of the following describes the impact of the above conditions, and the direction the US will give?

☐ **A** The Fire Suppression system is alarming as a warning of a potential for a discharge. Per TRM 3.7.9.4, "Halon Fire Suppression System", provide backup fire suppression and establish a fire watch, when the room has cleared.

☑ **B** The Fire Suppression system is alarming as a warning of a potential for a discharge. Per TRM 3.3.3.7, "Fire Detection Instrumentation", the Zone 45 fire detection system is inoperable and a fire watch must be established.

☐ **C** The Fire Suppression system is warning that a discharge will occur after a timer countdown. Per TRM 3.7.9.4, "Halon Fire Suppression System", provide backup fire suppression and establish a fire watch, when the room has cleared.

☐ **D** The Fire Suppression system is warning that a discharge will occur after a timer countdown. Per TRM 3.3.3.7, "Fire Detection Instrumentation", the Zone 45 fire detection system is inoperable, establish a fire watch, when the room has cleared.

Justification

B - CORRECT; The East and West DC switchgear rooms require two zones (one photoelectric smoke detector and one ion smoke detector) to initiate a halon release. Activation of one smoke detector zone, ion or photoelectric, will cause the strobe and horn to pulse slowly. However, the TRM requires all detectors to be functioning or the system is inoperable

A - WRONG; The Halon system is not made inoperable because the detection system has a failure.
Plausible; Examinee may think that due to the "false" activation of a sensor, the system should be prevented from any subsequent activation and the Halon system can no longer trigger.

C - WRONG; Activation of a second smoke detector of the opposite type, but in the same room, will cause the strobe and horns for the affected room to pulse QUICKLY. The flashing lights will operate, and a 60 second pre-discharge time delay will begin. Upon expiration of the time delay the Halon System will discharge and the strobe and horn will sound steadily.
Plausible; Examinee may think that the SLOWLY pulsating horn and strobe light warn of a timer countdown to discharge halon, in which case, the Halon system would then be inoperable and this action would be correct.

D - WRONG; Only one detector failing in the activate mode would cause the given alarms.
Plausible; Examinee may think that the pulsating horn and strobe lights indicate that the failed detector has caused a full system malfunction and a discharge is imminent. If the system were actually triggered due to multiple detector failures, this would be the correct choice.

References

1. OP 2341A, "Fire Protection System", Pg 4, Discussion section.
2. ARP 2590I, "Alarm Response for Fire Panel, C-26" (Zone 45), Pg 67-69

Comments and Question Modification History

Bob K. - D-3/W (Change choices to say "per" applicable procedure and make correct answer reference applicable ARP).
Modified question to utilize procedures where applicable action is directed. - RLC.
Bill M. - D-3/C, K
Angelo - D-3/C; No comments.

NRC K/A System/E/A **System** 086 Fire Protection System (FPS)

Number A2.03 **RO** 2.7 **SRO** 2.9 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following mal- functions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent actuation of the FPS due to circuit failure or welding

Question #: 18	Question ID: 9000011	☐ RO	☑ SRO	☐ Student Handout?	☐ Lower Order?
I-SRO Ques. # 18	Rev. 0	☑ Selected for Exam		Origin: New	☐ Past NRC Exam?

Setpoint:

ZONE 45

WEST DC SWGR ROOM FLP-6

- Trouble – Detector string failure
- Fire –
  - Detector string alarm
  - Halon release from pressure switch

AUTOMATIC FUNCTIONS

1. Activation of one smoke detector (photoelectric or ion) causes the following to occur:
   - Alarmed detectors red light illuminates.
   - On FLP-6, alarmed zone annunciates.
   - Locally strobe and horns pulses slowly.
   - On FLP-6, detectors location is shown on graphic annunciator.
   - Alarm signal is sent to C-26 Zone 45.
   - Alarm lamp for graphic annunciator for east 120 volt switchgear room illuminates.
2. Activation of another smoke detector of opposite type (ion or photoelectric), causes the following to occur:
   - Alarmed detectors red light illuminates.
   - On FLP-6, alarmed zone annunciates.
   - Locally strobe and horns pulses quickly.
   - The following closes:
     - 2-HV-138VB, "Supply to 'B' Battery Room From DC Rooms"
     - 2-HV-601B, "Cable Vault to 'B' DC Room SWGR. Room Fire Damper"
   - 60 second pre-discharge time delay begins.
   - Upon expiration of time delay Halon System discharges and strobe and horn sounds steadily.
   - Flashing lights operates.
   - Upon expiration of time delay halon system discharges into west 120 volt switchgear room.

ZONE 45

STOP **WARNING** STOP

When Halon Systems are actuated, the affected area is neither oxygen deficient nor toxic; however, extended exposure to Halon may have harmful effects.

NOTE

Activation of the manual pull station causes a halon release after a five second time delay. An abort switch can be used to prevent a halon release until the affected panel can be reset. If the abort switch is turned back before the panel is reset, the Halon discharges after a ten second time delay. The reset is inside the respective FLP and a valve lock key is required to get into the panel. The manual pull station activation overrides the abort.

CORRECTIVE ACTIONS

1. Refer To Attachment 6, "FLP–6 and 6A Zone 45" and DETERMINE cause of alarm.
2. IF fire alarm is valid, PERFORM the following: ④
   - 2.1 DETERMINE location of fire.
   - 2.2 IF alarm was cause by actuation of Halon Fire Suppression System AND fire is verified, PERFORM the following:
     - EVACUATE affected area.
     - Refer To AOP 2559, "Fire" and PERFORM applicable actions.
   - 2.3 IF alarm is due to Halon discharge AND *no* fire is present, PERFORM the following:

CAUTION

When ventilating care must be taken not to discharge products of combustion into non–affected rooms.

   - 2.3.1 PERFORM actions to ventilate affected area.
   - 2.3.2 Refer To OP 2341A, "Fire Protection System," and REMOVE appropriate Halon System from service.

ZONE 45

2.3.3 POST fire watch as necessary.

2.3.4 NOTIFY Fire Marshall.

2.3.5 SUBMIT Trouble Report to Maintenance Department.

3. IF alarm is due to electrical malfunction, SUBMIT Trouble Report to Electrical Maintenance Department.
4. For continued operation, CONSIDER supplemental room cooling.
5. As applicable, Refer To Technical Requirements Manual, and DETERMINE system operability.

SUPPORTING INFORMATION

1. Initiating Devices
   - FPL-6
     - Detector string, FSD-49
       - 3 ion detectors (smoke)
       - 3 photoelectric detectors (smoke)
     - PS-7696
     - HS-7696 A & B (Manual Electric Release)
2. Computer Points
   - FLP-6
   - TE8436
3. Technical Requirements Manual, Section II, subsection 1.0,
   - table A.3.1..4
   - E.3.1
4. Procedures
   - OP 2341A, "Fire Protection System"
   - AOP 2559, "Fire"
   - AOP 2579F, "Fire Procedure for Hot Standby Appendix "R" Fire Area R-10"
   - AOP 2579FF, "Fire Procedure for Cooldown and Cold Shutdown Appendix "R" Fire Area R-10 and R-8"

Question #: 18 | Question ID: 9000011 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 18 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

TECHNICAL REQUIREMENTS

3/4.3 INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3 3.7 As a minimum, the fire detection instrumentation for each fire detection zone in TRM Table 3.3-10 shall be OPERABLE.

APPLICABILITY:

Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number of OPERABLE requirements of TRM Table 3.3-10:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour unless the instrument(s) is located inside the containment. Roving fire watches must monitor the area of the device in question, as

TECHNICAL REQUIREMENT
TABLE 3.3-10
FIRE DETECTION INSTRUMENTS

	Instrument Location (Zone)	Heat Total No. of Channels	Heat Minimum Channels Operable	Smoke Total No. of Channels	Smoke Minimum Channels Operable
4.	4.16 & 6.9 kV Switchgear Room (56'6") (40)	--	--	4	3
	4.16 & 6.9 kV Switchgear Room (31'6") (18)	--	--	4	3
	480 V West Switchgear Room (36'6") (18)	--	--	2	1
	480 V East Switchgear Room (36'6") (28)	--	--	2	1
	East DC Equipment Room (43 Alarm) (FLP 5)	--	--	6	6
	West DC Equipment Room (45 Alarm) (FLP 6)	--	--	6	6

Question #: **19** | ***Question ID:* 9000013** | ☐ **RO** | ☑ **SRO** | ☑ **Student Handout?** | ☐ **Lower Order?**

I-SRO Ques. # **19** | **Rev. 0** | ☑ **Selected for Exam** | ***Origin:* *New*** | ☐ **Past NRC Exam?**

While operating at Beginning of Life (BOL), 100% power, the "A" Main Feed Pump high vibration annunciator alarmed. After subsequent investigation and troubleshooting, the System Engineer and Maintenance agree that the pump must be removed from service within the next 60 minutes to prevent severe damage. The crew has just entered AOP 2575, Rapid Downpower.

Which of the following statements describes the method that must be utilized to perform this evolution?

☐ **A** Use Reactivity Plan RE-G-08 to reduce power to 70%, secure the Feed Pump, then transition to OP 2208, Attachment 5, Reactivity Thumbrules, for maintaining power with rising Xenon.

☐ **B** Use Reactivity Plan RE-G-05 until 90% power, then transition to AOP 2575, Attachment 7, Boration/Power Reduction Rates to continue the power reduction required to secure the Feed Pump.

☑ **C** Use Reactivity Plan RE-G-11 to reduce power to the appropriate level, secure the Feed Pump, then raise power to the appropriate level using OP 2204, Load Changes, and a new Reactivity Plan.

☐ **D** Use Reactivity Plan RE-G-10 to reduce power to the appropriate level, secure the Feed Pump, then transition to OP 2393, Core Power Distribution and Monitoring, to maintain ASI.

Justification

C IS CORRECT: Reactivity Plan RE-G-11 provides the Boration rate and CEA insertion required to down power to 55% within the one hour time constraint. OP 2322, Main Feedwater, requires power to be at or less than 55% to remove the first Main Feed Pump from service. This reactivity plan also provides the dilution rate to maintain power at 55% to compensate for Xenon. OP 2322, Main Feedwater, provides guidance to raise power as high as 75% once the affected Main Feed Pump is secured. As a result, a new reactivity plan must be developed to allow raising power.

A is incorrect; Reactivity Plan RE-G-08 only provides Boration rate and CEA Insertion required to downpower to 70%. Although OP 2208, Attachment 5, provides guidance for maintaining power with changing reactivity conditions (e.g., Xenon), AOP 2575 provides direction for use of a reactivity plan appropriate for the evolution being performed. Additionally, power must be at or below 55% to remove a Main Feed pump from service.
Plausible because this guidance is adequate to reduce power to 70%. OP 2322 provides guidance on continued operation with one Main Feed Pump up to 75% power. Examinee may think that a feed pump can be removed from service at less than or equal to 75%.

B is incorrect; Reactivity Plan RE-G-05 only provides Boration rate and CEA Insertion required to downpower to 90%. Although AOP 2575, Attachment 7, provides guidance for reducing power if a reactivity plan is not available, guidance exists for use of a reactivity plan appropriate for the evolution being performed.
Plausible because this guidance is adequate to remove the feed pump from service, but does not provide the complete guidance for reactivity control after the pump is secured.

D is incorrect; Reactivity Plan RE-G-10 provides guidance on reducing power to 55%; however, the reduction rate will NOT allow the Feed Pump to be removed from service in the required 60 minute time limit. OP 2393 provides guidance for maintaining ASI control with CEAs, but this guidance does NOT take into consideration the amount of Boric Acid to used to counteract Xenon.
Plausible because this reactivity plan will allow performance of the downpower to remove the Main Feed Pump from service, but it does NOT allow reducing power fast enough to meet the 60 minute time limit. The examinee may mistakenly assume that the reactivity plan does NOT include the effects of Xenon; therefore, the power reduction rate would actually be higher than indicated on RE-G-10.

References **Provided**

RE-G-03, Rapid Down Power Reactivity Plans (Question Reference #1 of 3)
AOP 2575, Rapid Downpower (NOT provided to examinees)

Comments and Question Modification History

Bob K. - D-3/W ("Bad question, "D" could be correct. Reword stem to say "within the next 60 minutes" and reword "C" & "D" to include 2204 and new reactivity plan).
Swapped Distaractor "A" and "B". Added OP 2204 to the correct answer (C.). Changed reactivity Plan to RE-G-10 instead of RE-G-14. (RE-G-14 may be appropriate as long as the crew stops at the required power level. Deleted OP 2204 and inserted OP 2393. RJA
Bill M. - D-2/C, K (Doesn't feel that B is plausible. No recommendation.) **Discussed. No modification needed. Distractors will allow the required downpower; therefore, all are plausible. - RJA**
Angelo - D-4/C; No comments.

NRC K/A System/E/A **System** 2.1 Conduct of Operations

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Question #: **19** I-SRO Ques. # **19** **Question ID: 9000013** ☐ RO ☑ SRO ☑ Student Handout? ☐ Lower Order?
Rev. 0 ☑ Selected for Exam **Origin: New** ☐ Past NRC Exam?

Number 2.1.43 RO 4.1 SRO 4.3 **CFR Link** (CFR: 41.10 / 43.6 / 45.6)

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

	RWST – 1 pump	**RWST** – 2 pump	**BAST** – 1%/min
90% ○	RE-G-04 BA vol - 880 gals BA flow - 44 gpm CEA pos - 180 steps	RE-G-05 BA vol - 880 gals BA flow - 88 gpm CEA pos - 180 steps	RE-G-06 BA vol - 204 gals BA flow - 20 gpm CEA pos - 180 steps
	Time - 20 min.	Time - 10 min.	Time - 10 min.
	Rate - 30%/hr	Rate - 60%/hr	Rate - 60%/hr
70% (No CEAs) ●	RE-G-07 BA vol - 2458 gals BA flow - 44 gpm CEA pos - 180 steps	RE-G-08 BA vol - 2640 gals BA flow - 88 gpm CEA pos - 180 steps	RE-G-09 BA vol - 606 gals BA flow - 20 gpm CEA pos - 180 steps
	Time - 56 min.	Time - 30 min.	Time - 30 min.
	Rate - 32%/hr	Rate - 60%/hr	Rate - 60%/hr
55% ●	RE-G-10 BA vol - 3080 gals BA flow - 44 gpm CEA pos - 154 steps	RE-G-11 BA vol - 3315 gals BA flow - 88 gpm CEA pos - 154 steps	RE-G-12 BA vol - 720 gals BA flow - 18 gpm CEA pos - 154 steps
	Time - 70 min.	Time - 38 min.	Time - 40 min.
	Rate - 39%/hr	Rate - 71%/hr	Rate - 68%/hr
15% ●	RE-G-13 BA vol - 4400 gals BA flow - 44 gpm CEA pos - 140 steps	RE-G-14 BA vol - 5120 gals BA flow - 88 gpm CEA pos - 140 steps	RE-G-15 BA vol - 1100 gals BA flow - 16 gpm CEA pos - 140 steps
	Time - 100 min.	Time - 58 min.	Time - 80 min.
	Rate - 51%/hr	Rate - 88%/hr	Rate - 64%/hr

Reactivity plans are provided and should be used if the initial conditions specified in the plans (100% initial power, ARO, desired final power level) are approximately as specified. These plans are an approximation of the required boration and CEA movement required to reach the desired power while controlling the ASI oscillation. The reactivity plan CEA positioning will maintain ASI within COLR limits. OP-2393, "Core Power Monitoring Distribution and Control," provides direction for maintaining ASI control within specified bands during steady-state conditions, transient conditions, or at the direction of Reactor Engineering. The reactivity plans provide the above Reactor Engineering direction for ASI control. ASI control during the down power in accordance with the reactivity plan is prefered, however, it should not interfere with event mitigation. Once reactor power is stabilized, ASI should be maintained in accordance with OP 2393 or the reactivity plan.

The first page of the reactivity plan provides the boration rate to initiate the down power and the desired CEA position for ASI control. The second page of the plan contains more detailed information for stabilizing the plant at the desired power level. This page should not be interpreted as procedural direction and deviation from this guidance is allowable to achieve the desired power level within the desired time. The third page provides a prediction of the relative ASI trend during the down power. The ASI trend should not be used as an indication of a true absolute ASI value. If at the completion of the down power, it was noted that significant deviation from the plan was required to achieve the desired power level, reactor engineering should be promptly informed.

Core Reactivity affects from RCS temperature changes vary significantly over core life, based upon RCS boron concentration and resultant Moderator Temperature Coefficient (MTC) value. At 100% power, beginning of life (BOL), after xenon equilibrium, the value of MTC is much less negative than at the end of life (EOL). This means that at BOL, a change of 1°F RCS temperature will cause approximately a $^1/_2$% change in power, whereas a 1°F RCS temperature change at EOL will cause approximately a 2% change in power.

1.1 Applicability

This procedure is applicable in Mode 1 at power levels greater than 20% when an emergency power reduction is required.

Question #: 19 | Question ID: 9000013 | ☐ RO | ☑ SRO | ☑ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 19 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

Millstone Unit 2 Rapid Downpower	AOP 2575 Revision 004-01 Page 7 of 35

INSTRUCTIONS	CONTINGENCY ACTIONS

CAUTION

In the case of a dropped CEA, rod motion is *not* used to initiate downpower.

___3.4 IF *not* downpowering due to a dropped rod, INSERT Group 7 CEAs 10 ± 2 steps to initiate downpower.

___3.5 Refer To PPC or Reactor Engineering Curve and Data Book and OBTAIN reactivity plan for the inital reactor power condition and desired load reduction.

3.5.1 IF reactor is *not* at the reactivity plan initial conditions, Refer To Attachment 7, and DETERMINE desired rate of load reduction for time in core life.

___3.6 IF desired to borate from the RWST (preferred method) PERFORM the following:

a. ENSURE at least one charging pump operating.

b. ENSURE CH-196, VCT makeup bypass, closed.

c. ENSURE CH-504, RWST to charging suction, open.

d. OPEN CH-192, RWST isolation.

d.1 IF CH-192, RWST isolation, can *not* be opened, Go To step 3.8.

e. CLOSE CH-501, VCT outlet isolation.

f. CHECK charging flow at desired rate.

f.1 START additional charging pumps as needed and balance charging and letdown.

g. Go To step 3.9.

___3.7 Based on required rate of downpower, START additional charging pumps as necessary and balance charging and letdown..

Level of Use Continuous

STOP THINK ACT REVIEW

SRO Exam Questions Only (No "Parents" Or "Originals")

Question #: 20	**Question ID:** 9000014	☐ RO	☑ SRO	☐ Student Handout?	☑ Lower Order?	
I-SRO Ques. # 20	Rev. 0	☑ Selected for Exam		**Origin:** New	☐ Past NRC Exam?	

Which of the following actions require authorization by the Refueling SRO?

☐ **A** When the grapple will NOT disengage the top of a fuel assembly, snap or twang the hoist cable to release the grapple.

☐ **B** When an overload occurs, use the hand crank on the refuel machine hoist to free the fuel assembly from the guide pins.

☐ **C** In an emergency, insert a fuel assembly into the core and ungrapple it provided NO other fuel assemblies are adjacent.

☑ **D** If an underload occurs prematurely, raise the fuel assembly, pull the mast detent pin, rotate slightly, and reinsert the assembly.

Justification

D IS CORRECT; OP 2209A, Attachment 3, provides a listing of SRO responsibilities. Included is statement that requires authorization from the Refuel SRO to perform various action contained in Attachment 4. Under "Difficulty Inserting", the Refueling SRO is responsible for authorizing the Mast Detent Pin to be pulled. If necessary, he/she may also authorize a slight rotation of the mast to allow inserting a fuel assembly.

A is incorrect. A Caution in Attachment 4 states, "Snapping or twanging the hoist cable is prohibited." However, the cable may be manipulated or pulled and gently released to eliminate a grapple hang-up.
Plausible because the examinee may be confused about what actually constitutes a hoist cable manipulation.

B is incorrect; While Attachment 4 allows the Refuel SRO to authorize several actions to free a fuel assembly (seen as an overload), manually manipulating the cable hoist is NOT one of them. The refuel machine may be moved in either horizontal plane to free an assembly.
Plausible because Attachment 4 allows the Refuel SRO to authorize movement of the refuel machine manually (hand crank) in either horizontal direction, just NOT in the vertical direction.

C is incorrect; A fuel assembly cannot be left unsupported, even in an emergency.
Plausible because, in an emergency, the SRO may authorize a fuel bundle to be inserted into any open (unsupported) location in the core; however, the grapple must remain attached.

References

OP 2209A, Refueling Operations

Comments and Question Modification History

Bob K. - D-5/C (Operator involvement in refuel operations is very limited. Question is still acceptable).
Bill M. - D-4/W, G (Didn't know due to limited involvement in fuel movement. Question is acceptable.)
Angelo - D-5/W; Not an operator task.

NRC K/A System/E/A **System** 2.1 Conduct of Operations

Generic K/A Selected

NRC K/A Generic **System** 2.1 Conduct of Operations

Number 2.1.35 **RO** 2.2 **SRO** 3.9 **CFR Link** (CFR: 41.10 / 43.7)
Knowledge of the fuel-handling responsibilities of SROs.

Attachment 3
Personnel Responsibilities During Refueling Operations
(Sheet 5 of 6)

5. Refuel SRO

- Authority
  - Stop CORE ALTERATIONS when deemed necessary.
  - Stop or defer any activity around the refuel floor which would jeopardize the safety of personnel or equipment.
- Responsibilities
  - Be present on the Refuel Floor and responsible for maintaining OPS Procedures as required during the following CORE ALTERATIONS:
    - Fuel Shuffle
    - Moving/replacing sources
    - CEA shuffle in the reactor vessel
    - Removing the upper guide structure from the reactor vessel ③
    - Uncoupling of CEA extension shafts
    - Removal of CEA extension shafts from the UGS
    - Recoupling of CEA extension shafts
    - Any other CORE ALTERATION as determined by Reactor Engineering
  - As necessary, the Refuel SRO following consultation with Reactor Engineering, authorizes that specified guidelines on Attachment 4 be performed by refueling personnel.
  - General monitoring responsibilities include but are not limited to the following:
    - Ensure the Exclusion Area around the refuel pool is maintained per MA–AA–102, "Foreign Material Exclusion" and DNAP–2000, "Dominion Work Management Process."
    - Ensure proper radiological practices are maintained around the refuel pool.
    - Ensure safe load paths per MP 2712B2, "Overhead Crane Operating Information," and MP 2712B1, "Control Heavy Load" are maintained.
    - Ensure general safety of personnel and equipment.
  - Ensure communications between refuel floor and the Control Room are maintained during CORE ALTERATIONS.
    - Halt CORE ALTERATIONS if communications are lost.
  - Ensure refueling equipment is operated in accordance with OPS–FH 215, "Refueling Machine Operation."

Attachment 4
Guidelines For Fuel Movement Operations
(Sheet 2 of 5)

CAUTION

Snapping or twanging of the hoist cable is prohibited.

- Hoist Cable Manipulation – Manipulation or pulling on the hoist cable is typically recommended only to assist in the following:
  - Allow engagement of the bottom nozzle with the core support plate guide pins.
  - Facilitate fuel movement when "hang–ups" or problems with grapple engagement or disengagement are encountered.
  - Facilitate fuel movement when potential grid interferences are encountered.
- Difficulty Inserting – IF inserting a fuel assembly into the core, SFP storage rack, fuel elevator, or upender can, AND an underload occurs, PERFORM the following:

1. RAISE the assembly until the underload is cleared.
2. CHECK alignment of fuel assembly and fixture.
3. IF repositioning a fuel assembly manually over the core, ENSURE spreader is raised.
4. As necessary, REPOSITION fuel assembly and TRY reinserting.
5. IF an underload is experienced again, PERFORM any of the following:

   5.1 PULL the RFM mast detent pin out and TRY reinserting.

   5.2 ROTATE the RFM mast slightly in the clockwise or counterclockwise direction and TRY reinserting.

   5.3 ENSURE fuel assembly is raised 4″ from the core support plate to clear the guide pins and HAND CRANK the RFM up to 0.3″ in any direction and TRY reinserting.

   5.4 IF the above does not clear underload, manipulate the hoist cable to free the fuel assembly from potential grid interferences.

Question #: **21** | **Question ID: 9000024** | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 21 | Rev. 0 | ☑ Selected for Exam | **Origin: New** | ☐ Past NRC Exam?

The plant is operating at 100% power when ISO New England and CONVEX operators notify Millstone Station that a "Degraded Voltage" condition exists. Voltage on the 4.16 kV buses is presently 3,900 volts.

Based on this information, which one of the following describes actions that the Unit Supervisor must direct, per the applicable procedures?

☐ **A** Rack out the 6.9 and 4.16 kV breakers to the RSST and slow-start both Emergency Diesel Generators.

☑ **B** Terminate surveillance testing of any safety related pumps and motors and secure them, if possible.

☐ **C** Commence a plant downpower and secure all unnecessary equipment as the lower power permits.

☐ **D** Ensure the "E" and "F" Instrument Air compressors are operating in the "Lead" and "Standby" modes.

Justification

B - CORRECT; To limit the risk of damage to safety related motor windings due to the higher current flows that would be expected, all unnecessary running of these components must be terminated.

A - WRONG; The RSST breakers are not disabled until voltage drops below 88.9% of rated voltage. The EDG are verified as running only if the RSST is in service.
Plausible; Examinee may believe that in order to prevent transferring to the RSST, which is getting power from a degraded grid voltage, the RSST breakers must be disabled so they will NOT close on a possible plant trip. Also, prestaging the EDGs with a "slow start" would put minimum stress on the machines which are destined to carry all plant AC loads. However, this action is premature and over conservative for the given conditions.

C - WRONG; The applicable AOP Does not direct a plant down power be commenced as the loss of power to the grid is a far worse impact than any gains by securing equipment.
Plausible; Examinee may believe that because the applicable AOP directs that unnecessary loads be secured to help with the degraded voltage, and a trip from a lower power level is preferred, that lowering power to allow securing of components is logical.

D - WRONG; The restart of the vital Instrument Air Compressors is handled by EOP-2525 post-trip. There is no benefit to prestaging their alignment as they must be locally "reset" on a loss of power regardless.
Plausible; Examinee may realize that on a probable trip from loss of the grid, Instrument Air recovery will require operator action, as the compressor that is normally running is not vitally powered. Therefore, in order to ensure Instrument Air remains available (and a Vital Auxiliary Safety Function is preserved), the prestaging of the IA compressor alignment is a logical action.

References

AOP-2580, Pg. 6, Step 3.1

Comments and Question Modification History

Bill M. - D-2/C, K
Angelo - D-4/C; No comments.

NRC K/A System/E/A **System** 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.17 **RO** 2.6 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Millstone Unit 2 Degraded Voltage	AOP 2580 Revision 003 – 04 Page 6 of 12

3.0 Degraded Voltage

INSTRUCTIONS	CONTINGENCY ACTIONS
3.1 IF surveillances of safety related pumps and motors are in progress, TERMINATE surveillances during degraded voltage conditions.	
3.2 REQUEST the SM refer to C OP 200.8, "Response to ISO NE/CONEX Emergencies and Alerts."	
* 3.3 CHECK actual degraded voltage condition exists by observation of **ANY** of the following conditions: • 4160 volt bus 24C OR 24D voltage less than 3,900 volts • 480 volt bus 22E OR 22F voltage less than 440 volts	

Level of Use
Continuous

STOP THINK ACT REVIEW

Question #: **22** | **Question ID: 9000015** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

I-SRO Ques. # **22** | **Rev. 0** | ☑ **Selected for Exam** | **Origin: New** | ☐ **Past NRC Exam?**

The Auxiliary Building PEO has just noted an active boric acid leak on the bottom of a flange on CH-198, "RCP Bleedoff Pressure Control Valve to VCT". The leak is very small (2-3 drops per minute), but boric acid deposits from the leak are corroding a pipe support bracket located below the flange.

Which of the following administrative control documents require action be taken to control this leak?

☑ **A** Final Safety Analysis Report, Chapter 15, License Renewal, Aging Management Programs

☐ **B** Technical Specifications, Reactor Coolant System Leakage, LCO 3.4.6.2

☐ **C** Technical Requirements Manual, Containment Isolation Valves, LCO 3.6.3.1

☐ **D** Operational Configuration Control, OP-AA-1500, Alternate Plant Configurations, Attachment 5

Justification

A - CORRECT; FSAR, Chapter 15, License Renewal, gives the requirements for a Boric Acid Corrosion control program. Specifically what a boric acid leak is corroding is not a factor in the need to control, stop and repair any problems caused by boric acid leakage.

B - WRONG; The given system degradation does NOT impact the Tech. Spec. for RCS leakage.
Plausible; RCP Bleedoff, which is part of the Letdown System, is connected directly to the RCS and has RCS water flowing through it; however, it is NOT considered RCS leakage because it can be isolated. Additionally, it cannot be considered pressure boundary leakage due to it's location.

C - WRONG; The TRM does not cover boric acid corrosion control and this small a leak would not effect the valve operability.
Plausible; Examinee may incorrectly believe that because this valve is part of the CTMT Isolation Specification in the TRM, that leakage from the flange would be covered here.

D - WRONG; Operation Configuration Control deals with control of plant system configuration based on component position changes, not configuration changes due to failures or degradation (corrosion) of system components.
Plausible; Examinee may think that if a component of a system cannot perform its function due to corrosion, then the system configuration must be affected.

References

FSAR, Chapter 15, License Renewal, Pg. 15-2 & 3; Boric Acid Corrosion Control

Comments and Question Modification History

Bob K. - D-2/C (Change leak to "letdown to clean RW system" and add admin procedures to all choices).
Changed location of leak to a flange on CH-198. Added specific sections or chapters to each distractor. RLC
Bill M. - D-2/W, K (Assumed the leak impacted the RCS Leakage Tech Spec. due to pressure boundary leakage. Discussed and realized flange leakage is NOT pressure boundary leakage. Additionally, this leak path is isolable; therefore, not RCS leakaage.)
Angelo - D-4/C; No comments.

NRC K/A System/E/A **System** 2.2 Equipment Control

Generic K/A Selected

NRC K/A Generic **System** 2.2 Equipment Control

Number 2.2.38 **RO** 3.6 **SRO** 4.5 **CFR Link** (CFR: 41.7 / 41.10 / 43.1 / 45.13)

Knowledge of conditions and limitations in the facility license.

15.2.1.3 Boric Acid Corrosion

Program Description

Boric Acid Corrosion corresponds to NUREG-1801, Section XI.M10 "Boric Acid Corrosion." The program manages the aging effect of loss of material and ensures that systems, structures, and components susceptible to boric acid corrosion are properly monitored. The program uses visual inspections to detect the boric acid leakage source, path, and any targets of the leakage. It ensures that boric acid corrosion is consistently identified, documented, evaluated, trended, and effectively repaired. The Boric Acid Corrosion program provides both detection and analysis of leakage of borated water inside containment. The General Condition Monitoring program is the primary method for detecting borated water leakage outside containment. The analysis of the leakage is performed through the Boric Acid Corrosion program. Any necessary corrective actions are implemented through the Corrective Action Program.

Boric Acid Corrosion program implements the requirements of:

- NRC Bulletin 2001-01 (Reference 15.2-15)
- NRC Bulletin 2002-01 (Reference 15.2-16)
- NRC Bulletin 2002-02 (Reference 15.2-17)
- NRC Bulletin 2003-02 (Reference 15.2-18)
- NRC Order EA-03-009 (Reference 15.2-19)
- NRC Bulletin 2004-01 (Reference 15.2-20)

The acceptance criterion is the absence of any boric acid leakage or precipitation. If boric acid leakage or precipitation is found by any personnel, it is required to be reported using the Corrective Action Program. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as part of the Quality Assurance Program. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable.

Question #: 22 | I-SRO Ques. # 22

Question ID: 9000015 ☐ RO ☑ SRO ☐ Student Handout? ☑ Lower Order?

Rev. 0 ☑ Selected for Exam Origin: New ☐ Past NRC Exam?

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-001

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

(MILLSTONE POWER STATION, UNIT NO. 2)

RENEWED FACILITY OPERATING LICENSE

Renewed License No. DPR-65

1. The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-65 issued on September 26, 1975 has now found that:

A. The application to renew License DPR-65 filed by Dominion Nuclear Connecticut, Inc. (DNC), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Part 54 Chapter 1, and all required notifications to other agencies or bodies have been duly made;

B. Construction of the Millstone Power Station, Unit 2, (facility) has been substantially completed in conformity with Construction Permit No. CPPR-76 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

C. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

D. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

E. There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

Question #: **23** | **Question ID: 8000044** | ☐ **RO** | ☑ **SRO** | ☐ **Student Handout?** | ☑ **Lower Order?**

I-SRO Ques. # **23** | **Rev. 2** | ☑ **Selected for Exam** | **Origin: Bank** | ☐ **Past NRC Exam?**

The Rad. Waste PEO has just brought an Aerated Radioactive Waste (ARW) Monitor Tank discharge permit to the Shift Manager for review and approval.
Upon reviewing the permit and ARW system status, the SM has noticed that the ARW monitor tank was sampled by chemistry for the generation of the discharge permit with a level of 85%. However, the tank now has an actual level of 95%.

Which of the following actions are required in order for the Shift Manager to approve discharging the ARW Monitor Tank?

☐ **A** Re-calculate the amount of the discharge based on the new tank volume, and note this on the existing discharge permit when complete.

☑ **B** Re-mix the tank for the required period of time, then resample the tank and generate a new discharge permit based on the new sample.

☐ **C** Re-sample the tank and generate a second discharge permit and discharge the tank based on the most conservative of the two permits.

☐ **D** Re-mix the tank contents to ensure thorough mixing with the previously sampled contents and discharge the tank on the existing permit.

Justification

B - CORRECT; The SM must verify, when approving the permit for release, that the ARW discharge is being done in "batch" mode. This means the tank contents must be a discrete quantity with a known level of radioactivity. Once the tank showed signs of additional input, the contents were no longer known. Therefore, the tank must be re-sampled before it could be legally discharged.

A - WRONG; The same discharge permit cannot be used, even if the tank volume is now known.
Plausible; This is actually what is done for every discharge to ensure the actual amount of the discharge is correctly documented. This must be done because the total amount listed on the permit assumes every gallon of tank volume will be discharged, but the discharge pump can NOT pump the tank down to zero and often trips off line with several percent left in the tank.

C - WRONG; Administrative requirements state that if the Rad. Monitor is operable it will be used per existing guidelines.
Plausible; This is what is done if the ARW discharge Rad. Monitor is NOT operable and the tank must still be discharged based on plant needs.

D - WRONG; This must be done before the tank is resampled for generation of a new permit.
Plausible; This is acceptable if the contents had been known (by sampling) but had stratified and was not indicating properly on the Discharge Rad Monitor.

References

REMODCM, Rad Waste Sampling Requirements (Batch Discharge).

Comments and Question Modification History

Bob K. - D-2/C (OK - possibly make more challenging by adding "recirc times" to choices)
No changes made.
Bill M. - D-2/C, K
Angelo - D-2/C; No comments.

NRC K/A System/E/A **System** 2.3 Radiation Control

Generic K/A Selected

NRC K/A Generic **System** 2.3 Radiation Control

Number 2.3.6 **RO** 2.0 **SRO** 3.8 **CFR Link** (CFR: 41.13 / 43.4 / 45.10)
Ability to approve release permits.

Question #: 23 | Question ID: 8000044 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☑ Lower Order?

I-SRO Ques. # 23 | Rev. 2 | ☑ Selected for Exam | Origin: Bank | ☐ Past NRC Exam?

Table I.C.-2
Millstone Unit 2 Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Source	Sample Type and Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection $(LLD)_A$ (µCi/ml)
A.Batch Release[B.]				
1.Clean Waste Monitor Tank, Aerated Waste Monitor Tank and Steam Generator Bulk[D.]	Grab sample prior to each batch release	Prior to each batch release	Principal Gamma Emitters[C.]	5×10^{-7}
			I-131	1×10^{-6}
			Ce-144	5×10^{-6}
2.Condensate Polishing Facility - Waste Neutralization Sump[E.]			Dissolved & Entrained Gases[K.]	1×10^{-5}
		Monthly Composite[F.,G.]	H-3	1×10^{-5}
		Quarterly Composite[F.,G.]	Gross alpha	1×10^{-7}
			Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B.Continuous Release				
1.Steam Generator Blowdown[H.] 2.Service Water Effluent[J.] 3.Turbine Sumps[L.] 4.RBCCW Sump[M.]	Daily Grab Sample[I.] & prior to aligning to Long Island Sound for RBCCW sump	Weekly Composite[F.,G.]	Principal Gamma Emitters[C.]	5×10^{-7}
			I-131	1×10^{-6}
			Ce-144	5×10^{-6}
	Monthly Grab Sample	Monthly	Dissolved & Entrained Gases[K.]	1×10^{-5}
	Weekly Grab or Composite	Monthly Composite[F.,G.]	H-3[N.]	1×10^{-5}
	Weekly Composite	Quarterly Composite[F.,G.]	Gross alpha	1×10^{-7}
			Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE I.C.-2
TABLE NOTATIONS

B. A batch release is the discharge of liquid wastes of a discrete volume from the tanks listed in this table. Prior to the sampling, each batch shall be isolated and at least two tank/sump volumes shall be recirculated or equivalent mixing provided. If the steam generator bulk can not be recirculated prior to batch discharge, samples will be obtained by representative compositing during discharge.

C. The LLD will be 5×10^{-7} μCi/ml. The principal gamma emitters for which this LLD applies are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6} μCi/ml. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level. When unusual circumstances result in <u>a priori</u> LLDs higher than required, the reasons shall be documented in the Radioactive Effluent Release Report.

D. <u>For the Steam Generator Bulk:</u>
<u>**IF**</u> the applicable batch gamma activity is not greater than 5×10^{-7} μCi/ml, <u>**THEN**</u> the sampling and analysis schedule for gross alpha, Sr-89, Sr-90, Fe-55 are not required.

STOP THINK ACT REVIEW

Question #: 24 | **Question ID: 9000016** ☐ RO ☑ SRO ☐ Student Handout? ☐ Lower Order?

I-SRO Ques. # 24 | Rev. 0 ☑ Selected for Exam **Origin: New** ☐ Past NRC Exam?

The plant was operating normally at 100% power when the crew manually tripped the plant due to a tube rupture on #2 Steam Generator. The crew successfully performed EOP 2525, Standard post Trip Actions, and entered EOP 2534, Steam Generator Tube Rupture.

The following conditions exist:

- SIAS, CIAS, and EBFAS have been verified.
- "A" and "B" RCPs are running with adequate NPSH.
- Main Steam Line Radiation Monitor, RM 4299C, are presently reading 1.5 R/hr and stable.
- Condenser Air Removal is aligned to the Unit 2 Stack.
- The crew is in the process of lowering both hot leg temperatures to less than or equal to 515°F.
- MSI has been overridden to maintain steam flow to the Condenser.
- The Unit 2 Stack Gaseous Radiation Monitor, RM 8132B, is in alarm reading 800 cpm and rising.

Which of the following statements describes the procedurally directed method used to limit the release of radiation to the environment?

☐ **A** Secure all Main Exhaust Fans and direct the Chemist to ensure the 95,000 microcurie/sec release limit will NOT be exceeded.

☐ **B** Ensure all flow from the Main Condenser to the Steam Jet Air Ejector Radiation Monitor, RM-5099, has been secured.

☐ **C** Override and start the remaining Main Exhaust Fans and ensure all Radwaste Ventilation supply fans are providing adequate flow.

☑ **D** Continue the cooldown and isolate #2 Steam Generator when both hot leg temperatures are less than or equal to 515°F.

Justification

D IS CORRECT; A continuing reading or 1.5 R/hr on the Main Steam Line Radiation Monitors, 4299C, is indication of fuel failure. With highly contaminated steam flowing to the Condenser, the Steam Jet Air Ejectors in service, Condenser Air Removal aligned to Unit 2 Stack, and a Main Exhaust Fan running, the release to the atmosphere will continue. This alignment must be maintained to allow cooldown of the affected S/G to the condenser. If the Atmospheric Dumps were used instead, the release to the environment would be considerably higher.

A is incorrect; Securing all Main Exhaust Fans would no longer allow Condenser Air Removal to remain in operation, resulting in a loss of vacuum and the inability to maintain steaming to the Main Condenser. Heat Removal would need to be established through the Atmospheric Dumps, which would result in a considerably higher release to the environment.
Plausible because the Annunciator Response for an alarm on the Unit 2 Stack Gaseous Radiation Monitor, RM 8132B, in the event of a LOCA is to stop all Main Exhaust Fans. Exceeding the release rate limit of 95,000 microcurie/sec is reportable. The unit is already in at least an Alert, Charlie-One due to the SGTR.

B is incorrect; The Steam Jet Air Ejector Radiation Monitor, RM-5099, monitors the air flow from the Main Condenser through the Condenser Air Removal System. Stopping this air flow would result in the loss of vacuum and the inability to maintain steaming to the Main Condenser. Heat Removal would need to be established through the Atmospheric Dumps, which would result in a considerably higher release to the environment.
Plausible because Condenser Air Removal Flow may be stopped which would limit the release of radiation, but only for a short period of time. (Loss of vacuum leads to the use of the Atmospheric Dumps.)

C is incorrect; Starting all Main Exhaust Fans and the Radwaste Ventilation Supply Fans will provide more dilution and lower the reading on the Unit 2 Stack Gaseous Radiation Monitor, RM 8132B; however, the actual release of radioactivity will still be the same. In fact it may be slightly higher in that short lived activity will be discharged sooner due to the increase in air flow.
Plausible because an examinee may recognize the lower reading on the Unit 2 Stack Gaseous Radiation Monitor, RM 8132B, as a decrease in the radioactive release rate.

References

EOP 2534, Steam Generator Tube Rupture and associated Technical Guide

Comments and Question Modification History

Bob K. - D-3/C (OK)
Bill M. - D-2/C, K
Angelo - D-3/C; **Changed Main Steam Line rad monitor to the Facility 2 side to match affected SG. - RLC**

NRC K/A System/E/A System 2.3 Radiation Control

Generic K/A Selected

Question #: 24 | Question ID: 9000016 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 24 | Rev. 0 | ☑ Selected for Exam | Origin: New | ☐ Past NRC Exam?

NRC K/A Generic System 2.3 Radiation Control

Number 2.3.14 **RO** 3.4 **SRO** 3.8 **CFR Link** (CFR: 41.12 / 43.4 / 45.10)

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Millstone Unit 2 Steam Generator Tube Rupture	**EOP 2534** **Revision 025-01** **Page 11 of 67**

INSTRUCTIONS	CONTINGENCY ACTIONS

Align Condenser Air Removal to Unit 2 Stack

*7. IF EBFAS has initiated AND the condenser is available, ALIGN the condenser air removal system to Unit 2 stack:

a. ENSURE condenser air removal fan, MF-55A or MF-55B, is running.

b. IF condenser air removal fan MF-55A is operating, ENSURE makeup damper, EB-171, is open.

c. OPEN EB-57, condenser air removal to Unit 2 stack.

d. ENSURE AC-11, Purge exhaust filter outlet damper is closed.

e. OPEN AC-59, Outside air makeup damper.

f. START ONE main exhaust fan.

g. ENSURE HV-118, Radwaste exhaust damper is closed.

h. START F-20, Fuel handling area supply fan.

i. ENSURE HV-173, Exhaust mod discharge damper is in "MOD" position.

j. PLACE AC-59, Outside air makeup damper to "MID" position.

STOP THINK ACT REVIEW

Step Number 7 Align Condenser Air Removal to Unit 2 Stack

The intent of this step is maintain condenser vacuum in order to use the steam dumps for an RCS cooldown. and to minimize the possibility of airborne activity backdrafting from the Millstone stack to Unit 2 after EBFAS has actuated.

INSTRUCTIONS

Following EBFAS, the Unit 2 Enclosure Building Ventilation is aligned to the Millstone stack, and isolates the Condenser Air Removal System from the Millstone stack. The lineup that exists does not provide for enough flow to maintain condenser vacuum. The isolation damper has a history of leakage which results in the Millstone stack gaseous releases backdrafting into Unit 2 if the condenser air removal fans are not operating. This step is designed to minimize the possibility of this occurring.

The operator is directed to ensure a condenser air removal fan is operating and to align a flow path to the Unit 2 Stack for the condenser air removal system following an EBFAS actuation. The operator must also ensure EB-171 is open if condenser air removal fan "A" is operating. This is due to the difference in size between the A and B fans. Additional ventilation fans are started and dampers are opened to balance the flow.

CONTINGENCY ACTIONS

None

JUSTIFICATION FOR DEVIATION

MP2 adds this plant specific step to ensure that a vacuum is maintained in the main condenser, if the main condenser is available. The use of the steam dumps is the preferred method to perform a plant cooldown. The steam dumps help maintain secondary inventory and during a SGTR will minimize offsite exposure.

REFERENCES

Safety Function Requirements Manual, Section 2.8.2.6.3 "Radiological Consequences of a Steam Generator Tube Rupture" (2/2000)

AR 99014319 Resolve Issue With Starting Main Exhaust Fans in EOP 2534

SRO Exam Questions Only (No "Parents" Or "Originals")

Question #: 25	Question ID: 9800061	☐ RO	☑ SRO	☐ Student Handout?	☐ Lower Order?
I-SRO Ques. # 25	Rev. 0	☑ Selected for Exam		Origin: Mod	☐ Past NRC Exam?

The plant is stable at 100% power with the Turbine Driven Auxiliary Feedwater Pump out of service for planned maintenance.

Then, the plant trips due to a loss of off site power (state wide blackout), resulting in the following conditions:

- The "A" Main Steam header ruptures in containment on the trip.
- Busses 24B and 24D are de-energized due to a bus fault on 24D.
- Facility One SIAS, CIAS, EBFAS, MSI and CSAS have all fully actuated.
- All feedwater has been secured to the #1 Steam Generator (SG).
- All other plant systems and components that have power are functioning as designed.

The crew is evaluating numerous alarms and indications caused by the power loss and subsequent ESD.

Which of the following alarm indications will require actions to be taken in EOP 2525?

☑ **A** C-05 alarms indicating a Excess Steam Demand on the #1 SG, and C-08 alarm indicating VR-21 is de-energized.

☐ **B** C-02/3 alarms indicating RCS temperatures are abnormally low and dropping, and both Boric Acid Pumps are de-energized.

☐ **C** C-05 alarms indicating both SG levels abnormally low, and only one Aux. Feedwater pump is feeding just the #2 SG.

☐ **D** C-01 alarms indicating CTMT Spray has actuated, and C-01 indicating only two CAR fans and one CS pump are operating.

Justification

A - CORRECT; All alarms and indications mentioned in the four choices are expected for the given event, a loss of the RSST and 24D, with a subsequent ESD on the "A" Main Steam header. VR-21 is deenergized based on the loss of 24D. This will prevent the "B" Atmospheric Dump Valve (ADV) from being operated from the control room (after about 10 minutes) and the ESD in the Enclosure building prevents local operation. Therefore, immediate action is required to get an operator to C21 (Remote Shutdown Panel) to control RCS temperature when the affected SG boils dry (thus preventing PTS).

B - WRONG; This gives indication of an excessive cooldown of the RCS with a potential problem with boric acid injection. However, the other facility of power is available to allow automatic alignment of a boric acid source to the remaining charging pump, which is sufficient (although not optimum) to meet "reactivity control".
Plausible; Procedure steps will give guidance to align additional boron injection, but this is above the required amount.

C - WRONG; Auxiliary Feedwater will feed enough to recover level in the unaffected steam generator, by design.
Plausible; The SG levels are abnormal compared to an uncomplicated trip due to volume shrinkage from the ESD. Procedures give guidance to start additional AFW pumps, as necessary (none available in this case) to return SG levels to the operating band (40-70%). However, the only available option with the given conditions is "Once-Through-Cooling", which can not be done in EOP-2525.

D - WRONG; One facility of CTMT Cooling and Pressure Control is certainly NOT optimum during and ESD, but it is designed to be sufficient to maintain CTMT Integrity, provided all feed is secured to the affected SG in the required time frame.
Plausible; Procedure gives guidance to repower and start all available ESAS components. However, this is not required to prevent exceeding a design limit.

References

EOP 2525, Step 17 and AOP 2501, Pg. 10, Diagnostic Chart

Comments and Question Modification History

Bob K. - D-4/C (OK - good question).
Bill M. - D-3/C, K
Mike C. - Modify stem to eliminate possibility of improving AFW condition in 2525. **- Done - RLC**
Angelo - D-3/C; **Added to stem that all feedwater is secured to #1 SG. - RLC**

NRC K/A System/E/A System 2.4 Emergency Procedure /Plan

Generic K/A Selected

NRC K/A Generic System 2.4 Emergency Procedures /Plan

Number 2.4.45 **RO** 4.1 **SRO** 4.3 **CFR Link** (CFR: 41.10 / 43.5 / 45.3 / 45.12)
Ability to prioritize and interpret the significance of each annunciator or alarm.

Millstone Unit 2 Standard Post Trip Actions	EOP 2525 Revision 023 Page 24 of 26

INSTRUCTIONS	CONTINGENCY ACTIONS
Subsequent Actions (continued)	
___17. IF steam generator pressure is less than 572 psia AND the most affected steam generator has boiled dry, as indicated by CET temperature rising, OPERATE the ADV for the least affected steam generator to stabilize CET temperature.	

Question #: 25 | Question ID: 9800061 | ☐ RO | ☑ SRO | ☐ Student Handout? | ☐ Lower Order?

I-SRO Ques. # 25 | Rev. 0 | ☑ Selected for Exam | Origin: Mod | ☐ Past NRC Exam?

Millstone Unit 2 Diagnostic for Loss of Electrical Power	AOP 2501 Revision 001 – 02 Page 10 of 10

Attachment 1
Lost Control Power
(Sheet 1 of 1)

Diagnostic Indication (deenergized)	Loss of VR–11	Loss of VR–21	Loss of VA–10	Loss of VA–20
	Left side C–01 & CEAPDS	Right side C–01 & Core Mimic	"A" safety channels & "A" RPS	"B" safety channels & "B" RPS
	PA System lost	RRS Battery B/U lasts for ~ 10 min. after VR–21loss.		
C–02	Letdown Isolates (High Temp sensor failure)	RRS battery B/U available PZR setpoint at program ** RRS battery depleted ** PZR setpoint fails to zero – Minimum Charging flow – Maximum Letdown flow	Chg Pp suction to RWST PZR Lvl control in Ch "X" – Maximum Charging flow – Minimum Letdown flow	PZR Lvl control in Ch "Y" – Maximum Charging flow – Minimum Letdown flow
C–03	Lose All PZR Heaters (Select in "X" or "X+Y")	Lose All PZR Heaters (Select in "Y" or "X+Y") PZR Backup Heaters All four banks unavailable	Lose All PZR Heaters (Select in "X" or "X+Y")	Lose All PZR Heaters (Select in "Y" or "X+Y")
C–03	B/U Charging Pumps Both start automatically (Dead control circuit)	** RRS battery depleted ** B/U Charging Pumps Both stop w/ PZR stpnt @ 0	HIC–100E dead ("E" spray) PZR Spray Auto Control unavailable. [Manual spray control w/ HIC–100F only]	HIC–100F dead ("F" spray) [Auto control w/ HIC–100E if Press control in Ch. "X"]
C–03	Loop 1 temp to RRS lost (RRS auto bypasses)	Loop 2 temp to RRS lost (RRS auto bypasses)	Loop 1 temp input to ICC, LTOP, RRS all lost (RRS auto bypasses)	Loop 2 temp input to ICC, LTOP, RRS all lost (RRS auto bypasses)
C–05		TIC 4165 Dead (Tavg cntrl)	PIC–4223 Dead (#1 ADV)	PIC–4216 Dead ("A" CDV) PIC–4224 Dead (#2 ADV)
C–05		Condenser Steam Dumps All four failed closed (Open w/ Quick Open only)		Condenser Steam Dumps PIC–4216 de–energized [Contrl w/ TIC–4165 @ C05 or PIC–4216 @ Foxboro IA]
		RRS battery B/U available Quick Open on program ** RRS battery depleted ** No QO to any dump valve		
C–05	#1 ADV (PIC–4223) avail. Manual only (Input pressure frozen when power lost)	RRS battery B/U available #2 ADV (PIC–4224) avail. (Input pressure frozen at value when power was lost). ** RRS battery depleted ** #2 ADV fails closed [Control from C21/C10]	#1 ADV remote control lost (C–05 & C–21 contrls dead) [Local–Manual only]	#2 ADV remote control lost (C–05, –21, –10 contrl dead) [Local–Manual only]
C–05	"A" SGFP Insert dark, controls still work (Indication on PPC)	"B" SGFP Insert dark, controls still work (Indication on PPC)	#1 FRV Main – Fail as is [Local–Manual only] (Closes on power restore)	#2 FRV Main – Fail as is [Local–Manual only] (Closes on power restore)
C–05	"A" SGFP Minimum Flow Cntrl dead, valve fail closed	"B" SGFP Minimum Flow Cntrl dead, valve fail closed	#1 FRV Bypass fails closed (All control is lost)	#2 FRV Bypass fails closed (All control is lost)
C–05	Blowdown isolates (B/D Rad Monitor dead)	Blowdown isolates (SJAE Rad Monitor dead)	#1 Aux FRV fails open [Local–Manual only]	#2 Aux FRV fails open [Local–Manual only]

Level of Use Information | STOP THINK ACT REVIEW

SRO EXAM Answer Key

Order Is Based on Sequence #	Stem ID - Rev		
# 1	9000018 - *0*	D	Direct the crew to commence a controlled cooldown and depressurization.
# 2	9000019 - *0*	B	The HPSI Pumps are showing signs of cavitation due to Containment Sump clogging. Secure both Containment Spray Pumps.
# 3	9000003 - *0*	A	The "A" RCP Excess Flow Check Valve has seated. Manually trip the reactor and turbine, then stop the "A" RCP.
# 4	9000004 - *0*	C	Dispatch a PEO to manually operate the "B" Auxiliary Feedwater Regulating Valve, 2-FW-43B. This will prevent excessive auxiliary feedwater from overfilling the <u>affected</u> SG.
# 5	9082581 - *0*	B	1. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the <u>clockwise</u> direction as directed by the Control Room. 2. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the <u>counterclockwise</u> direction as directed by the Control Room. 2-SI-657, SDC Heat Exchanger Flow Control Valve, must be opened first to establish the desired RCS cooldown rate.
# 6	3100002 - *2*	C	When pressure lowers to less than 80 psig. The loss of many important controls, such as Feedwater, could degrade plant conditions at the time of the trip; therefore, the reactor must be tripped when control of important systems could become challenged.
# 7	9000020 - *0*	D	CEA #1 Upper Electrical Limit must be overridden <u>and</u> the CMI must be bypassed for CEA recovery. <u>Only</u> the CMI will be INOPERABLE while the CEA is being recovered.
# 8	9000005 - *0*	D	Feed the affected SG to maintain level 40 to 45%. This will cover the SG tube to allow for Iodine scrubbing and still allow adequate volume to accept water from the tube rupture.
# 9	9000006 - *0*	C	Alert, Charlie-One

SRO EXAM Answer Key

Order Is Based on Sequence #	Stem ID - Rev		
# 10	9000007 - *0*	B	With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 $\leq$ 60 micro-curies/gram once per 4 hours. Operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 micro-curies/gram limit. The 4 hour sampling period allows time to obtain and analyze a sample. There is a low probability of a steam line break or S/G tube rupture in the next 48 hours and it is expected that normal coolant Iodine concentration would be restored within 48 hours.
# 11	9000008 - *0*	A	Immediately secure the "A" RCP, raise RCS pressure, then start "C" and "D" RCPs.
# 12	9600016 - *0*	A	Per the ARP for C-38; shift pressurizer heater control to channel "Y" and restore pressurizer heaters, to ensure adequate margin from DNB is maintained.
# 13	53730 - *4*	C	Trip the reactor and Go to EOP 2525, "Standard Post Trip Actions".
# 14	9000021 - *0*	A	All fuel movement in containment must remain secured.
# 15	9000010 - *0*	D	The "A" Service Water Pump is considered OPERABLE. If the "A" Service Water Strainer Flush Valve, 2-SW-90A, exceeds the Maximum "Normal Limit" on an immediate retest, then declare the "A" Service Water Pump inoperable.
# 16	9000012 - *0*	C	Ensure a control room operator is specifically assigned to close the Containment Purge Valves within 30 minutes of an event, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.
# 17	9079010 - *0*	B	Lower the setpoint on the "A" steam dump and log into the DNB Technical Specification.
# 18	9000011 - *0*	B	The Fire Suppression system is alarming as a warning of a potential for a discharge. Per TRM 3.3.3.7, "Fire Detection Instrumentation", the Zone 45 fire detection system is inoperable and a fire watch must be established.
# 19	9000013 - *0*	C	Use Reactivity Plan RE-G-11 to reduce power to the appropriate level, secure the Feed Pump, then raise power to the appropriate level using OP 2204, Load Changes, and a new Reactivity Plan.
# 20	9000014 - *0*	D	If an underload occurs prematurely, raise the fuel assembly, pull the mast detent pin, rotate slightly, and reinsert the assembly.

SRO EXAM Answer Key

Order Is Based on Sequence #	Stem ID - Rev		
# 21	9000024 - *0*	B	Terminate surveillance testing of any safety related pumps and motors and secure them, if possible.
# 22	9000015 - *0*	A	Final Safety Analysis Report, Chapter 15, License Renewal, Aging Management Programs
# 23	8000044 - *2*	B	Re-mix the tank for the required period of time, then resample the tank and generate a new discharge permit based on the new sample.
# 24	9000016 - *0*	D	Continue the cooldown and isolate #2 Steam Generator when both hot leg temperatures are less than or equal to 515°F.
# 25	9800061 - *0*	A	C-05 alarms indicating a Excess Steam Demand on the #1 SG, and C-08 alarm indicating VR-21 is de-energized.

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Question # 1

The plant automatically tripped on High Pressurizer Pressure due to an inadvertent closure of the Main Turbine Control Valves.

During the performance of EOP 2525, Standard Post Trip Actions, the crew reported that Bus 24D is deenergized due to a fault <u>and</u> that Power Operated Relief Valve (PORV), RC-404, is stuck open. All other equipment operated as designed. Upon entry into EOP 2532, Loss of Coolant Accident, the following conditions exist:

- Containment pressure is 4.5 psia and slowly rising.
- Reactor vessel is 43% and slowly going down
- CET temperatures are 578°F and stable
- RCS pressure is 1310 psia and stable
- Pressurizer level is 100%.
- Steam generator levels are both 41% and going up slowly.

Which of the following actions must the Unit Supervisor/Shift Manager perform to preserve a Safety Function?

A Direct the Technical Support Center to develop a plan to restore RWST level.

B Direct the Balance of Plant Operator to align Condenser Air Removal to the Unit 2 Stack.

C Direct the Reactor Operator to place the SI/CS Pump Miniflow switches in "OPERATE".

D Direct the crew to commence a controlled cooldown and depressurization.

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Question # 2

The plant tripped from 100% power due to a Large Break LOCA. The crew successfully completed all actions of EOP 2525, Standard Post Trip Actions, and are presently performing EOP 2532, Loss of Coolant Accident.

The following conditions exist approximately 2 hours after the trip:
* SRAS actuated approximately 15 minutes ago.
* Containment pressure is 5 psig and slowly lowering.
* RCS pressure is 360 psia and slowly lowering.
* CET temperatures indicate 434°F and slowly lowering.
* HPSI Pump current and flow are fluctuating.

Which of the following describes the cause of the HPSI Pump current and flow fluctuations, and the initial action that must be directed?

A Hot water in the Containment Sump is flashing to steam in the HPSI Pump suctions.
Start at least 2 CAR Fans in Fast speed.

B The HPSI Pumps are showing signs of cavitation due to Containment Sump clogging.
Secure both Containment Spray Pumps.

C Boron is beginning to plate out in the core causing alternately high and low HPSI flow.
Establish Hot Leg and Cold Leg Injection.

D Total Safety Injection flow is higher than necessary for the present conditions.
Throttle the HPSI Injection valves as needed.

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Question # 3

While operating at 100% power, the RCP A UPPER SEAL PRES HI annunciator alarms. While referring to the appropriate Annunciator Response Procedure, the RCP A BLEED-OFF FLOW HI annunciator alarms. Within a minute, the RCP A BLEED-OFF FLOW HI annunciator clears and the RCP A BLEED-OFF FLOW LO annunciator alarms and remains lit. Numerous annunciators associated with "A" RCP seals also alarm.

Which of the following describes the reason for this sequence of annunciators <u>and</u> the direction that must be given?

A The "A" RCP Excess Flow Check Valve has seated. Manually trip the reactor and turbine, then stop the "A" RCP.

B The "A" RCP Middle Seal has failed. Evaluate the condition of the other seals to confirm no other degradation or failures.

C The Bleedoff Pressure Controller, PIC-215, has malfunctioned. Using the Foxboro Controller, restore "A" RCP Bleedoff pressure and flow to the normal band.

D The RCP Bleedoff Relief Valve has inadvertently opened. Evaluate "A" RCP Seal pressures to determine whether or not the "A" RCP may remain in operation.

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Question # 4

The plant is operating at 100% power with the "B" Auxiliary Feedwater (AFW) Pump out of service for maintenance.

Then the following events occur:
* Automatic plant trip due to a Steam Generator Tube Rupture (SGTR) on #2 Steam Generator (SG).
* Loss of the RSST and VA-20 at the time of trip.
* Shortly after the trip, a Safety Injection Actuation signal (SIAS) automatically actuated.
* All other plant systems respond as designed.

Which one of the following actions <u>must</u> the US perform during EOP 2525, Standard Post Trip Actions, to mitigate the consequences of this event <u>and</u> what is the reason for this action?

A Dispatch a PEO to the Hot Shutdown Panel, C-21, to throttle open the #2 Atmospheric Dump Valve. This will permit a cooldown of <u>both</u> Hot Leg temperatures to ≤ 515°F.

B Direct the BOP to swap the control power supply switch for the Terry Turbine to Facility 1. This will allow the operator to maintain <u>both</u> S/G levels in the prescribed bands.

C Dispatch a PEO to manually operate the "B" Auxiliary Feedwater Regulating Valve, 2-FW-43B. This will prevent excessive auxiliary feedwater from overfilling the <u>affected</u> SG.

D Direct the BOP to close #2 S/G Steam Supply to the Terry Turbine, MS-202, after the disconnect is closed. This will minimize the radioactive release from the <u>affected</u> SG.

Question # 5

The plant has experienced a loss of VA-10 while in Mode 5 with Shutdown Cooling in operation.

Assuming RBCCW flow was NOT diverted from the SDC Heat Exchangers by any other system, which of the following actions would be performed outside the Control Room and what is the reason for performing these actions in the listed order?

A 1. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.
2. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.

2-SI-306, SDC Total Flow Control Valve, must be opened first to provide minimum flow for the operating LPSI Pump.

B 1. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.
2. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.

2-SI-657, SDC Heat Exchanger Flow Control Valve, must be opened first to establish the desired RCS cooldown rate.

C 1. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.
2. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.

2-SI-657, SDC Heat Exchanger Flow Control Valve, must be opened first to establish the desired RCS cooldown rate.

D 1. Place 2-SI-306, SDC Total Flow Control Valve, in manual and turn the handwheel in the clockwise direction as directed by the Control Room.
2. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve, in manual and turn the handwheel in the counterclockwise direction as directed by the Control Room.

2-SI-306, SDC Total Flow Control Valve, must be opened first to provide minimum flow for the operating LPSI Pump.

Question # 6

The plant is operating at 100% power when the Balance of Plant (BOP) operator reports that Instrument Air header pressure is at 95 psig and lowering. Immediately following, the Turbine Building PEO reports a large unisolable leak just downstream of the "D" Instrument Air Dryer After Filters.

Assuming Instrument Air header pressure continues to lower, at what pressure in the Instrument Air System must the Unit Supervisor (US) direct a manual reactor trip (by procedure) and why?

A Prior to reaching 85 psig.
When pressure drops below 85 psig the crew is procedurally directed to crosstie Station Air with Unit 3. Operation in this alignment will result in all components supplied by Instrument Air being inoperable, which is an unanalyzed condition.

B When pressure lowers to less than 85 psig.
At approximatley 85 psig the Instrument Air/Station Air Crosstie valve opens. Continued operation with Station Air supplied to valves and controllers will result in erratic operation of components due to the high moisture content of Station Air.

C When pressure lowers to less than 80 psig.
The loss of many important controls, such as Feedwater, could degrade plant conditions at the time of the trip; therefore, the reactor must be tripped when control of important systems could become challenged.

D Prior to reaching 80 psig.
The Auxiliary Feed Regulating Valves will lock up with less than 80 psig supply pressure. The reactor must be tripped to allow the initial automatic opening of these valves and begin feeding Steam Generators.

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Question # 7

The reactor is at 100% power with the CEA Motion surveillance in progress. When Group 7 CEA #1 is tested, CEAPDS indicates it inserts two steps, then slips an additional 20 steps. The appropriate actions were taken to stabilize RCS temperature and the following conditions were observed:

* Reactor power stable at ~ 99%.
* Only Upper Electrical Limit lights are energized on the core mimic.
* CEA #1 indicates 158 steps withdrawn on CEAPDS.
* CEA #1 indicates 178 steps withdrawn on the PPC.
* CEA Motion Inhibit (CMI) alarms on C-04
* CEAPDS Group Deviation indication for CEA #1

Fifty (50) minutes after CEA #1 slipped, all required actions per AOP-2556, CEDS Malfunctions, have been completed, including plant power changes.
Also, I&C reports the circuit malfunction that caused CEA #1 to slip has been repaired and the CEA can now be recovered.

Which one of the following describes actions that must be taken to recover CEA #1 and what is the administrative concern of those actions?

A Pulse counts must be reset to clear the Upper Core Stop and the CMI must be bypassed for CEA recovery. CEA #1 Pulse Count Indication and the CMI will be INOPERABLE while the CEA is being recovered.

B CEA #1 Upper Electrical Limit must be overridden and the CMI must be bypassed for CEA recovery. Reed Switch Indication for CEA #1 and the CMI will be INOPERABLE while the CEA is being recovered.

C Pulse counts must be reset to clear the Upper Core Stop and the CMI must be bypassed for CEA recovery. Only the CMI will be INOPERABLE while the CEA is being recovered.

D CEA #1 Upper Electrical Limit must be overridden and the CMI must be bypassed for CEA recovery. Only the CMI will be INOPERABLE while the CEA is being recovered.

Senior Reactor Operator NRC License Upgrade Exam

Question # 8

The reactor is manually tripped from 100% power due to a Steam Generator Tube Rupture (SGTR). On the trip, the RSST is lost due to grid instabilities. All other systems respond normally. EOP 2525, Standard Post Trip Actions, is entered.

What is the direction for controlling the affected SG level during the performance of EOP 2525, Standard Post Trip Actions, <u>and</u> what is the basis for this direction?

A Secure all feedwater to the affected SG. This will provide the maximum volume to accept water from the tube rupture and still allow a cooldown to 515°F to isolate the affected SG.

B Maintain the affected SG level 40 to 70%. This will maintain the SG tube covered to allow a cooldown to 515°F and still maintain adequate volume to accept water from the tube rupture.

C Maintain at least 300 gpm feedwater to the affected SG for Heat Removal. The addition of clean water will provide dilution of radioactivity which will lower the release to the environment.

D Feed the affected SG to maintain level 40 to 45%. This will cover the SG tube to allow for Iodine scrubbing and still allow adequate volume to accept water from the tube rupture.

Senior Reactor Operator NRC License Upgrade Exam

Question # 9

The plant is in MODE 5 performing a normal cooldown for refueling. "B" LPSI Pump is in service supplying both SDC Heat Exchangers. RCS to SDC Temperature, T351X, is presently reading 187°F with RCS pressure being held at 150 psia.

Suddenly, Bus 24D is deenergized due to a fault. Fifteen minutes after the loss of Bus 24D, the following conditions are reported:

- RCS pressure is 155 psia and slowly rising.
- RCS to SDC Temperature, T351X, is reading 186°F and stable.
- CET temperatures are 205°F and slowly rising.
- RVLMS indicates vessel level at 100%.
- Both S/G levels are 60% and stable
- Containment is being evacuated.

NO other operator actions have been taken.

Which of the following notifications must be made?

A General Interest, Echo

B Unusual Event, Delta-One

C Alert, Charlie-One

D Site Area Emergency, Charlie-Two

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Question # 10

An RCS chemistry sample taken at 100% power indicates 6 micro-curies/gram DOSE EQUIVALENT I-131.

Which of the following describes the required action and the basis for that action?

A With the specific activity of the primary coolant > 0.1 micro-curies/gram DOSE EQUIVALENT I-131, be in COLD SHUTDOWN within 36 hours after detection. Isotopic analysis of the primary coolant must be performed once per hour when activity of the primary coolant > 0.1 micro-curies/gram DOSE EQUIVALENT I-131.
The hourly sampling period allows time to obtain and analyze a sample. There is a low probability of a steam line break or S/G tube rupture in the next 36 hours and there is significant conservatism built into the RCS specific activity limit.

B With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 $\leq$ 60 micro-curies/gram once per 4 hours. Operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 1.0 micro-curies/gram limit.
The 4 hour sampling period allows time to obtain and analyze a sample. There is a low probability of a steam line break or S/G tube rupture in the next 48 hours and it is expected that normal coolant Iodine concentration would be restored within 48 hours.

C With the specific activity of the primary coolant > 0.1 micro-curies/gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours from the time of detection and in COLD SHUTDOWN within 48 hours from the time of detection.
The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and prevent exceeding the radiological release limit at the site boundary from an assumed LOCA.

D With the specific activity of the primary coolant > 1.0 micro-curies/gram DOSE EQUIVALENT I-131, lower the RCS specific activity to $\leq$ 1.0 micro-curies/gram DOSE EQUIVALENT I-131 within the next 36 hours or be in HOT STANDBY within the following 6 hours.
It is expected that normal coolant Iodine concentration would be restored within 36 hours. If not, adequate time is provided to achieve HOT STANDBY to prevent exceeding Control Room dose limits from an assumed LOCA.

Question # 11

A plant heatup has just been started and the following conditions presently exist:

- RCS Temperature is at 205°F and slowly rising.
- RCS pressure is stable at the minimum allowed for "A" and "B" RCP operation.
- "A" and "B" RCPs have just been started.
- Shutdown Cooling has just been secured.
- "C" and "D" RCP breakers have just been racked up.

Then, the "B" RCP trips when the breaker's overcurrent relay actuated due to being jarred while moving staging (NOT an actual overcurrent condition).

Which of the following actions are required under the present conditions?

A Immediately secure the "A" RCP, raise RCS pressure, then start "C" and "D" RCPs.

B Immediately place Shutdown Cooling back in operation, then secure the "A" RCP.

C Immediately start "C" and "D" RCPs, then secure the "A" RCP.

D Immediately start the "C" RCP and operate it with the "A" RCP.

Question # 12

A Rapid Downpower at 50%/hr is in progress due to an RCS leak in containment that exceeds the administrative limit. The following plant conditions presently exist:

* Plant power is 93% and dropping at the intended rate.
* Pressurizer level is 65% and stable.
* RCS pressure is 2250 psia and stable.
* One charging pump is running, Letdown is at approximately 30 gpm.
* Adding boric acid to the charging pump suction to maintain the desired rate of power reduction.
* Forcing Pressurizer Sprays in progress.
* C02/3 annunciator in alarm; D-37, "PZR PRESSURE SELECTED CHANNEL DEVIATION HI/LO"
* Channel "Y" Pressurizer Level and Pressure controlling normally.

Then, during the load reduction, Pressurizer Level Channel "X" fails to zero (0) and the following occur:

* All control systems respond as designed to the failure.
* C02/3 annunciator in alarm; A-38, "PRESSURIZER CH X LEVEL HI/LO".
* C02/3 annunciator in alarm; C-38, "PRESSURIZER CH X LEVEL LO-LO".
* PPC alarms on Monitor #2 indicative of the instrument failure.

Which of the following actions must the Unit Supervisor direct <u>and</u> why?

A Per the ARP for C-38; shift pressurizer heater control to channel "Y" and restore pressurizer heaters, to ensure adequate margin from DNB is maintained.

B Per SP-2602A RCS Leakage; deselect Pressurizer Level Channel "X" from the leak rate calculation, to ensure valid trending of RCS Leak Rate by the PPC.

C Per the ARP for A-38; place the standby charging pumps in "Pull-To-Lock", to prevent the rate of the plant downpower from accelerating above 50%/hr.

D Per AOP-2575, Rapid Downpower; shift pressurizer heater control to channel "Y" and restore the pressure controller setpoint, to prevent a plant trip on TM/LP.

Question # 13

A reactor startup is in progress using CEA withdrawal. The RO has just stopped withdrawing Group # 7 CEAs and makes the following announcements:

* The reactor is critical.
* Startup rate is positive and stabilizing at ~1.5 DPM.

Which of the following actions should the Reactivity SRO direct?

A Commence Emergency Boration until the reactor is subcritical.

B Insert the Group #7 CEAs to lower the startup rate below 0.5 DPM.

C Trip the reactor and Go to EOP 2525, "Standard Post Trip Actions".

D Insert all CEAs per OP-2206, "Reactor Shutdown" and notify RE.

Question # 14

The plant is in Mode 6 with the following conditions:

* Core re-load in progress and approximately half way completed.
* "A" LPSI pump running for Shutdown Cooling (SDC) operation.
* "B" LPSI pump in standby, aligned for SDC use.
* "A" train of Spent Fuel Pool (SFP) cooling in service.

Then, "A" LPSI pump is lost due to a breaker fault. When the "B" LPSI pump is started, it seizes and trips on breaker overload.
The Unit Supervisor (US) then directs the RO to recover SDC using the "B" Containment Spray (CS) pump.

Which of the following additional directions must the US give, while SDC flow is being supplied by a CS pump?

A All fuel movement in containment must remain secured.

B SDC supplementing of SFP cooling must be secured.

C Containment must remain evacuated of non-essential personnel.

D Containment Closure must be fully set with all access doors closed.

Senior Reactor Operator NRC License Upgrade Exam

Question # 15

The required surveillance must be performed after repairs were made to the "A" Service Water Strainer Flush Valve, 2-SW-90A. The completed surveillance indicates the valve stroke time is slightly above the Maximum 'Normal Limit", but is below the Maximum "Acceptable Limit". All other parameters are within the "Acceptable Limits".

Which of the following describes the condition of the "A" Service Water Pump <u>and</u> the required action?

A The "A" Service Water Pump remains inoperable. Perform the required repairs to the "A" Service Water Strainer Flush Valve, 2-SW-90A, then perform the required surveillances to restore the "A" Service Water Pump to OPERABLE.

B The "A" Service Water Pump remains inoperable. Obtain a different set of test equipment and immediately retest the "A" Service Water Strainer Flush Valve, 2-SW-90A, again to verify that the previous data was accurate.

C The "A" Service Water Pump is considered OPERABLE. Place the "A" Service Water Pump Strainer in an Augmented Testing Program and test weekly to ensure the "A" Service Water Strainer Flush Valve, 2-SW-90A, remains OPERABLE.

D The "A" Service Water Pump is considered OPERABLE. If the "A" Service Water Strainer Flush Valve, 2-SW-90A, exceeds the Maximum "Normal Limit" on an immediate retest, then declare the "A" Service Water Pump inoperable.

Senior Reactor Operator NRC License Upgrade Exam

Question # 16

The plant is in MODE 6 with the following conditions:

- Fuel movement is in progress.
- The Personnel Airlock Doors are open
- The Equipment Hatch is open.
- Containment Purge is in operation.
- Containment Atmosphere Radiation Monitor, RM-8123, is out of service for repairs.

The Auxiliary Building PEO has just reported that the blower for Containment Atmosphere Radiation Monitor, RM-8262, has tripped and is very hot to the touch.

Which of the following actions must be taken <u>and</u> why?

A Immediately suspend CORE ALTERATIONS and establish Containment Closure prior to resuming fuel movement, to ensure a potential fuel handling accident in Containment is NOT released to the environment.

B Immediately suspend CORE ALTERATIONS and restore the Radiation Monitor blower prior to resuming fuel movement, to ensure a potential fuel handling accident in Containment is NOT released to the environment.

C Ensure a control room operator is specifically assigned to close the Containment Purge Valves within 30 minutes of an event, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.

D Restore the Containment Purge Valves to OPERABLE status within the next 30 minutes or immediately close the Purge Valves, to ensure Containment Closure is reestablished in case of a fuel handling accident in Containment.

Question # 17

The plant was at 100% power when CONVEX ordered Main Generator output be lowered from 900 MWe to 600 MWe in 15 minutes.
AOP 2557, "Emergency Generation Reduction", was initiated and the following conditions now exist:

* Group 7 CEAs are at 170 steps withdrawn.
* Main Generator output is 610 MWe and slowly lowering.
* "A" Steam Dump Bypass Valve is 75% open and stable.
* "BYPASS TO CND", PIC-4216 output is 83% and stable.
* "B", "C" and "D" Steam Dump Bypass Valves are open 75% and stable.
* "STEAM DUMP TAVG CNTL", TIC-4165 output is 85% and stable.
* "RC LOOP 1 COLD LEG TEMP HI" annunciator has just alarmed (C02/3; C-34).
* "RC LOOP 2 COLD LEG TEMP HI" annunciator has just alarmed (C02/3; D-34).
* RCS Tcold is 550 °F and slowly rising (RPS).

Which one of the following actions should the US direct?

A Transfer control of the steam dumps to Foxboro IA control and lower Tcold to program.

B Lower the setpoint on the "A" steam dump and log into the DNB Technical Specification.

C Immediately trip the reactor and go to EOP 2525, "Standard Post Trip Actions".

D Insert CEAs until Tcold is back on program and both C02/3 alarms have cleared.

Question # 18

The plant is in normal operation at 100% power, when a Fire System Trouble annunciator is received on C06/7 and Zone 45 on Fire Panel, C-26. The Auxiliary Building PEO subsequently calls from the West DC Switchgear Room and reports the following:

* One Ion Chamber smoke detector is in alarm.
* The Halon strobe lights and horn are pulsating slowly.
* All other smoke detectors are operating normally (not in alarm).
* There is no smoke or fire in the area. The detector appears to have failed.

Which of the following describes the impact of the above conditions, and the direction the US will give?

A The Fire Suppression system is alarming as a warning of a potential for a discharge. Per TRM 3.7.9.4, "Halon Fire Suppression System", provide backup fire suppression and establish a fire watch, when the room has cleared.

B The Fire Suppression system is alarming as a warning of a potential for a discharge. Per TRM 3.3.3.7, "Fire Detection Instrumentation", the Zone 45 fire detection system is inoperable and a fire watch must be established.

C The Fire Suppression system is warning that a discharge will occur after a timer countdown. Per TRM 3.7.9.4, "Halon Fire Suppression System", provide backup fire suppression and establish a fire watch, when the room has cleared.

D The Fire Suppression system is warning that a discharge will occur after a timer countdown. Per TRM 3.3.3.7, "Fire Detection Instrumentation", the Zone 45 fire detection system is inoperable, establish a fire watch, when the room has cleared.

Question # 19

While operating at Beginning of Life (BOL), 100% power, the "A" Main Feed Pump high vibration annunciator alarmed. After subsequent investigation and troubleshooting, the System Engineer and Maintenance agree that the pump must be removed from service within the next 60 minutes to prevent severe damage. The crew has just entered AOP 2575, Rapid Downpower.

Which of the following statements describes the method that must be utilized to perform this evolution?

A Use Reactivity Plan RE-G-08 to reduce power to 70%, secure the Feed Pump, then transition to OP 2208, Attachment 5, Reactivity Thumbrules, for maintaining power with rising Xenon.

B Use Reactivity Plan RE-G-05 until 90% power, then transition to AOP 2575, Attachment 7, Boration/Power Reduction Rates to continue the power reduction required to secure the Feed Pump.

C Use Reactivity Plan RE-G-11 to reduce power to the appropriate level, secure the Feed Pump, then raise power to the appropriate level using OP 2204, Load Changes, and a new Reactivity Plan.

D Use Reactivity Plan RE-G-10 to reduce power to the appropriate level, secure the Feed Pump, then transition to OP 2393, Core Power Distribution and Monitoring, to maintain ASI.

Question # 20

Which of the following actions require authorization by the Refueling SRO?

A When the grapple will NOT disengage the top of a fuel assembly, snap or twang the hoist cable to release the grapple.

B When an overload occurs, use the hand crank on the refuel machine hoist to free the fuel assembly from the guide pins.

C In an emergency, insert a fuel assembly into the core and ungrapple it provided NO other fuel assemblies are adjacent.

D If an underload occurs prematurely, raise the fuel assembly, pull the mast detent pin, rotate slightly, and reinsert the assembly.

Senior Reactor Operator NRC License Upgrade Exam

Question # 21

The plant is operating at 100% power when ISO New England and CONVEX operators notify Millstone Station that a "Degraded Voltage" condition exists. Voltage on the 4.16 kV buses is presently 3,900 volts.

Based on this information, which one of the following describes actions that the Unit Supervisor must direct, per the applicable procedures?

A Rack out the 6.9 and 4.16 kV breakers to the RSST and slow-start both Emergency Diesel Generators.

B Terminate surveillance testing of any safety related pumps and motors and secure them, if possible.

C Commence a plant downpower and secure all unnecessary equipment as the lower power permits.

D Ensure the "E" and "F" Instrument Air compressors are operating in the "Lead" and "Standby" modes.

Senior Reactor Operator NRC License Upgrade Exam

Question # 22

The Auxiliary Building PEO has just noted an active boric acid leak on the bottom of a flange on CH-198, "RCP Bleedoff Pressure Control Valve to VCT". The leak is very small (2-3 drops per minute), but boric acid deposits from the leak are corroding a pipe support bracket located below the flange.

Which of the following administrative control documents require action be taken to control this leak?

A Final Safety Analysis Report, Chapter 15, License Renewal, Aging Management Programs

B Technical Specifications, Reactor Coolant System Leakage, LCO 3.4.6.2

C Technical Requirements Manual, Containment Isolation Valves, LCO 3.6.3.1

D Operational Configuration Control, OP-AA-1500, Alternate Plant Configurations, Attachment 5

Senior Reactor Operator NRC License Upgrade Exam

Question # 23

The Rad. Waste PEO has just brought an Aerated Radioactive Waste (ARW) Monitor Tank discharge permit to the Shift Manager for review and approval.
Upon reviewing the permit and ARW system status, the SM has noticed that the ARW monitor tank was sampled by chemistry for the generation of the discharge permit with a level of 85%. However, the tank now has an actual level of 95%.

Which of the following actions are required in order for the Shift Manager to approve discharging the ARW Monitor Tank?

A Re-calculate the amount of the discharge based on the new tank volume, and note this on the existing discharge permit when complete.

B Re-mix the tank for the required period of time, then resample the tank and generate a new discharge perm t based on the new sample.

C Re-sample the tank and generate a second discharge permit and discharge the tank based on the most conservative of the two permits.

D Re-mix the tank contents to ensure thorough mixing with the previously sampled contents and discharge the tank on the existing permit.

Senior Reactor Operator NRC License Upgrade Exam

Question # 24

The plant was operating normally at 100% power when the crew manually tripped the plant due to a tube rupture on #2 Steam Generator. The crew successfully performed EOP 2525, Standard post Trip Actions, and entered EOP 2534, Steam Generator Tube Rupture.

The following conditions exist:

- SIAS, CIAS, and EBFAS have been verified.
- "A" and "B" RCPs are running with adequate NPSH.
- Main Steam Line Radiation Monitor, RM 4299C, are presently reading 1.5 R/hr and stable.
- Condenser Air Removal is aligned to the Unit 2 Stack.
- The crew is in the process of lowering both hot leg temperatures to less than or equal to 515°F.
- MSI has been overridden to maintain steam flow to the Condenser.
- The Unit 2 Stack Gaseous Radiation Monitor, RM 8132B, is in alarm reading 800 cpm and rising.

Which of the following statements describes the procedurally directed method used to limit the release of radiation to the environment?

A Secure all Main Exhaust Fans and direct the Chemist to ensure the 95,000 microcurie/sec release limit will NOT be exceeded.

B Ensure all flow from the Main Condenser to the Steam Jet Air Ejector Radiation Monitor, RM-5099, has been secured.

C Override and start the remaining Main Exhaust Fans and ensure all Radwaste Ventilation supply fans are providing adequate flow.

D Continue the cooldown and isolate #2 Steam Generator when both hot leg temperatures are less than or equal to 515°F.

Senior Reactor Operator NRC License Upgrade Exam

Question # 25

The plant is stable at 100% power with the Turbine Driven Auxiliary Feedwater Pump out of service for planned maintenance.

Then, the plant trips due to a loss of off site power (state wide blackout), resulting in the following conditions:

- The "A" Main Steam header ruptures in containment on the trip.
- Busses 24B and 24D are de-energized due to a bus fault on 24D.
- Facility One SIAS, CIAS, EBFAS, MSI and CSAS have all fully actuated.
- All feedwater has been secured to the #1 Steam Generator (SG).
- All other plant systems and components that have power are functioning as designed.

The crew is evaluating numerous alarms and indications caused by the power loss and subsequent ESD.

Which of the following alarm indications will require actions to be taken in EOP 2525?

A C-05 alarms indicating a Excess Steam Demand on the #1 SG, and C-08 alarm indicating VR-21 is de-energized.

B C-02/3 alarms indicating RCS temperatures are abnormally low and dropping, and both Boric Acid Pumps are de-energized.

C C-05 alarms indicating both SG levels abnormally low, and only one Aux. Feedwater pump is feeding just the #2 SG.

D C-01 alarms indicating CTMT Spray has actuated, and C-01 indicating only two CAR fans and one CS pump are operating.