

# ANSWER KEY

ES-401

Site-Specific SRO Written Examination  
Cover Sheet

Form ES-401-8

<b>U.S. Nuclear Regulatory Commission</b>  <b>Site-Specific SRO Written Examination</b>	
<b>Applicant Information</b>	
Name: <i>Answer key</i>	
Date: <i>01/29/10</i>	Facility/Unit: <i>Millstone/Unit 2</i>
Region:    I <input checked="" type="checkbox"/> II <input type="checkbox"/> III <input type="checkbox"/> IV <input type="checkbox"/>	Reactor Type: W <input type="checkbox"/> CE <input checked="" type="checkbox"/> BW <input type="checkbox"/> GE <input type="checkbox"/>
Start Time:	Finish Time:
<b>Instructions</b>	
<p>Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.</p>	
<b>Applicant Certification</b>	
<p>All work done on this examination is my own. I have neither given nor received aid.</p> <p style="text-align: right; margin-right: 100px;"><i>Answer key</i> _____ Applicant's Signature</p>	
<b>Results</b>	
RO <del>SRO-Only</del> Total Examination Values	<i>N/A</i> / ____ / <u>25</u> Points
Applicant's Scores	<i>N/A</i> / ____ / <u>25</u> Points
Applicant's Grade	<i>N/A</i> / ____ / <u>100</u> Percent

# ANSWER KEY

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

Question #: 1

Question ID: 9000018

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 1

Rev. 1

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 and 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 568°F and stable
- RCS pressure is 1210 psia and stable
- Pressurizer level is 100%.
- RWST level is 96% and slowly going down.
- 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 24E to receive power from Unit 3.
  - 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 HPSI 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).

Plausible 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 required ONLY if the LOCA is outside of Containment.

B is incorrect; Although the loss of 24D makes the "C" HPSI pump unavailable, the one available HPSI pump should be enough to mitigate the event, provided an RCS cooldown and depressurization is accomplished.

Plausible the examinee may believe that starting a second HPSI pump is necessary to recover vessel level because the given conditions indicate that SI flow is presently inadequate (vessel level going down).

C is incorrect; The SI/CS Pump Miniflow switches are not placed in 'OPERATE' until RWST level is  $\leq 20\%$ .

Plausible 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

NRC (comments on original question) - Distracter "B" does not relate to a Safety Function; Replace. Distracter "A" would be acceptable if an RWST level were added to the stem.

RLC - In the stem: changed CET temp. from 578°F to 568°F, changed RCS pressure from 1310 psia to 1210 psia, and added "RWST level is 96% and slowly going down.". Also, changed choice "B" from realigning Cond. Air Removal to aligning 24E to Unit 3. [12/30/09] Bruce F. - D-3/C, No comment

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

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

Question #: **1**

Question ID: **9000018**

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # **1**

Rev. **1**

Selected for Exam

Origin: **New**

Past NRC Exam?

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.

## 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<sub>C</sub> or CET temperatures where RCS has stabilized.
4. T<sub>C</sub> 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<sub>C</sub> greater than 220° F the cooldown rate is 100° F/hr.
2. RCS T<sub>C</sub> less than or equal to 220° F the cooldown rate is 50° F/hr.

#### Perform Controlled Cooldown

\*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

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

Question #: 1

Question ID: 9000018

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 1

Rev. 1

Selected for Exam

Origin: New

Past NRC Exam?

- 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).

Level of Use  
Information

STOP

THINK

ACT

REVIEW

OP 2260

Rev. 009-03

13 of 60

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

Question #: 2

Question ID: 9000019

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 2

Rev. 1

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 precipitation is beginning to occur causing intermittent blockage of flow channels. 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 and blocking core flow.

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

NRC - (original question comment) Distracter "C" poor. Rework to "Boron precip. is beginning to occur causing intermittent blockage flow channels."

RLC - Reworked Choice "C" per NRC comments.

Bruce F. - D-4/C, No comment

**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

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

Question #: **2**

Question ID: **9000019**

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # **2**

Rev. **1**

Selected for Exam

Origin: **New**

Past NRC Exam?

**Millstone Unit 2  
Loss of Coolant Accident**

**EOP 2532**

**Revision 029**

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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

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

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 and 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 in 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. Plausible 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. Plausible 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.

Plausible 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.

Bruce F. - D-3/C, No comment

**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.

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

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.

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

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  $\leq 515^{\circ}\text{F}$ .
  - 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 overflowing 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  $<515^{\circ}\text{F}$ , 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

Bruce F. - D-4/C, No comment

**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.

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

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?

## 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.

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

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?

**Millstone Unit 2  
Standard Post Trip Actions**

EOP 2525

Revision 023

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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%.

d. CHECK that RCS subcooling is greater than or equal to 30° F.

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.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.

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

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?

**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

(continue)

- b.1 **IF** 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) **IF** 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)

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

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?

## Millstone Unit 2 Emergency Operating Procedure Technical Guide

## EOP 2525, Standard Post Trip Actions

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### 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 overflow.

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 overflow 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.

Bruce F. - D-5/C, Is it important to remember that these valves are reverse operating?

# 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?

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**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.

# 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?

## 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

# 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?

## Millstone Unit 2 Loss of Shutdown Cooling

AOP 2572

Revision 009-04

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### 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)

# 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?

## Millstone Unit 2 Loss of Shutdown Cooling

AOP 2572

Revision 009-04

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### 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:
- ESTABLISH communications between operators at the valve ("A" ESF Room) and Control Room.
  - CLOSE instrument air supply for SI-306, SDC total flow control.
  - OPEN petcock on instrument air supply pressure regulator and VENT SI-306, SDC total flow control, valve operator.
  - UNLOCK and REMOVE chain from manual handwheel.
  - ROTATE manual handwheel *counterclockwise* and ALIGN holes in outer shaft with hole in inner shaft.
  - INSERT the pin into shaft holes.
  - ENSURE SI-306, SDC total flow control, valve position indicator on the manual actuator is at throttled open position.
  - 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.

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

Question #: 6

Question ID: 310002

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 cross tie 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 approximately 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.

Bruce F. - D-3/C, No comment

**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

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

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?

## Millstone Unit 2 Loss of Instrument Air

AOP 2563

Revision 009-07

Page 3 of 32

### 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.

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

Question #: 7

Question ID: 9000020

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 7

Rev. 1

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%.
- \* Only the Upper Electrical Limit lights for all CEAs 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 defeated by lifting a lead 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 defeated by lifting a lead and the CMI must be bypassed for CEA recovery.  
Only 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 not 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.

Plausible; 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.

Plausible; 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.

Plausible; 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

Bruce Ferguson & Cliff Chapin - Change defeating the UEL to the actual action taken by the procedure.

Modified choices "B" & "D" (correct answer) per reviewer comment. - RLC 12/22/09

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

Question #: 7

Question ID: 900020  RO  SRO  Student Handout?  Lower Order?

I-SRO Ques. # 7

Rev. 1  Selected for Exam Origin: New  Past NRC Exam?

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

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

Question #: 7

Question ID: 900020

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 7

Rev. 1

Selected for Exam

Origin: New

Past NRC Exam?

## 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.

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

Question #: 7

Question ID: 900020

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 7

Rev. 1

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).

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

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 70%.

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 - SG level range in Choice "B" justification does not match range in distracter.**

RLC - Changed explanation in choice "B" from "40 to 75%" to "40 - 70%", to match distracter. [11/13/09]

Bruce F. - D-2/C, No comment

**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.

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

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?

## 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].

Level of Use  
Information

STOP

THINK

ACT

REVIEW

OP 2260

Rev. 009-03

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# 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.  
Bruce F. - D-3/C, No comment

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

# 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?

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 Fl., 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

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

Question #: 10

Question ID: 9000007

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 10

Rev. 1

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 > 1.0 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 > 1.0 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 > 1.0 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 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.

### 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  $\leq$  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  $\leq$  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  $\leq$  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  $\leq$  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

NRC - Remove the numbers (" $>0.1$ " and " $>1.0$ ") from all choices or make them all the same.

RLC - Changed " $> 0.1$ " in Choices "A" and "C" to " $> 1.0$ ". [11/30/09]

Bruce F. - D-4/W, 50/50 (C and D)

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

Question #: 10

Question ID: 9000007

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 10

Rev. 1

Selected for Exam

Origin: New

Past NRC Exam?

**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

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

Question #: **10**

Question ID: **900007**

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # **10**

Rev. **1**

Selected for Exam

Origin: **New**

Past NRC Exam?

October 27, 2008

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

- a.  $\leq 1.0 \mu\text{Ci}/\text{gram}$  DOSE EQUIVALENT I-131, and
- b.  $\leq 1100 \mu\text{Ci}/\text{gram}$  DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, 4.

ACTION:

- a. With the specific activity of the primary coolant  $> 1.0 \mu\text{Ci}/\text{gram}$  DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131  $\leq 60 \mu\text{Ci}/\text{gram}$  once per 4 hours.
- b. With the specific activity of the primary coolant  $> 1.0 \mu\text{Ci}/\text{gram}$  DOSE EQUIVALENT I-131 but  $\leq 60 \mu\text{Ci}/\text{gram}$ , operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the  $1.0 \mu\text{Ci}/\text{gram}$  limit. Specification 3.0.4 is not applicable.
- c. With the specific activity of the primary coolant  $> 1.0 \mu\text{Ci}/\text{gram}$  DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or  $> 60 \mu\text{Ci}/\text{gram}$  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  $> 1100 \mu\text{Ci}/\text{gram}$  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  $1100 \mu\text{Ci}/\text{gram}$  limit. Specification 3.0.4 is not applicable.
- e. With the specific activity of the primary coolant  $> 1100 \mu\text{Ci}/\text{gram}$  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.

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

Question #: **11**

Question ID: **9000008**

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # **11**

Rev. **1**

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 is the correct response 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 is 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

NRC - (original question comments) Reword stem question from "actions are required under" to "actions is the correct response under"

RLC - Reworded stem per NRC comments. [11/30/09]

Bruce F. - D-4/C, No comment

**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.

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

Question #: **11**

Question ID: **9000008**

RO  SRO

Student Handout?

Lower Order?

I-SRO Ques. # **11**

Rev. **1**

Selected for Exam

Origin: **New**

Past NRC Exam?



**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)

Level of Use  
**Continuous**

STOP      THINK      ACT      REVIEW

OP 2201

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

Question #: **11**  
 I-SRO Ques. # **11**

Question ID: **9000008**    RO    SRO    Student Handout?    Lower Order?  
 Rev. **1**    Selected for Exam   **Origin: New**    Past NRC Exam?

**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 (10)
      - MODE 4, TSAS 3.4.1.3 ACTION c
      - 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:
    - MODE 3, TSAS 3.4.1.2 ACTION b (10)
    - 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

Level of Use <b>C</b> ontinuous
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STOP      THINK      ACT      REVIEW

OP 2201  
 Rev. 031-12  
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# SRO Exam Questions Only (No "Parents" Or "Originals")

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 and 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 all PZR heaters. The heater control switch must be selected to ignore the failed channel and the heater breakers must be manually reclosed (they will not auto close even though the automatic controls are calling for more heater output). If this 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.  
Plausible: 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.  
Plausible: 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: All 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.  
Plausible: 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 distractor 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.

Bruce F. - D-3/C; No comment

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

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?

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

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

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

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?

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

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Rev. 000  
Page 1 of 1

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

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-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.

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Rev. 001-00  
Page 3 of 4

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

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 tripping 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.  
Plausible: 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.  
Plausible: Correct action if SUR briefly 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".  
Plausible: 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.  
Bruce F. - D-2/C, No comment

**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.

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

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?

## 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%  $\Delta Q$  (0.9%  $\Delta Q$  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.

Level of Use  
Continuous

STOP

THINK

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OP 2202

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

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.

Bruce F. - D-4/C, 50/50 (A and B)

**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.

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

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?

## Millstone Unit 2 Loss of Shutdown Cooling

AOP 2572

Revision 009-04

Page 3 of 70

### 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.

Level of Use  
Continuous

STOP

THINK

ACT

REVIEW

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

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?

**Millstone Unit 2  
Loss of Shutdown Cooling**

AOP 2572

Revision 009-04

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INSTRUCTIONS

CONTINGENCY ACTIONS

5.10 (continued)

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

- 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 <i>not</i> 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)

Level of Use  
**Continuous**

STOP

THINK

ACT

REVIEW

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

Question #: **14**  
 I-SRO Ques. # **14**

Question ID: **9000021**     RO     **SRO**     Student Handout?     Lower Order?  
 Rev. **0**     Selected for Exam    **Origin: New**     Past NRC Exam?

**Millstone Unit 2**  
**Loss of Shutdown Cooling**

**AOP 2572**                      **Revision 009-04**  
**Page 31 of 70**

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.

**Level of Use**  
**Continuous**

**STOP      THINK      ACT      REVIEW**

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

Question #: **15**

Question ID: **9000010**

RO  SRO

Student Handout?

Lower Order?

I-SRO Ques. # **15**

Rev. **0**

Selected for Exam

Origin: **New**

Past NRC Exam?

The required surveillance is being performed on the "A" Service Water Strainer Flush Valve, 2-SW-90A. The data indicates the valve stroke time is slightly above the Maximum 'Normal Limit', but is within the Operational Readiness Results (aka; Acceptable Limits).

Which of the following describes the condition of the "A" Service Water Strainer Flush Valve, 2-SW-90A and the required action?

- .....
- A** The "A" Service Water Strainer Flush Valve, 2-SW-90A is considered 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 Strainer Flush Valve, 2-SW-90A to OPERABLE.
  - B** The "A" Service Water Strainer Flush Valve, 2-SW-90A is considered inoperable. Obtain a different set of test equipment and immediately retest the "A" Service Water Strainer Flush Valve, 2-SW-90A, to verify that the previous data was accurate.
  - C** The "A" Service Water Strainer Flush Valve, 2-SW-90A is considered OPERABLE. Place the "A" Service Water 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 Strainer Flush Valve, 2-SW-90A is considered OPERABLE. If the "A" Service Water Strainer Flush Valve, 2-SW-90A, falls outside the Normal Limits on an immediate retest, then the "A" Service Water Strainer Flush Valve, 2-SW-90A is 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 valve 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 due to a failure on the first test.

Plausible 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.

Plausible 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 valve to be considered OPERABLE. The Service Water valve may be tested on a more frequent basis; however, the stroke time must be performed immediately after the initial failure to ensure it meets the "Normal" limit.

Plausible because the examinee may not be aware of the need to immediately perform a second stroke 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.

Bruce F. - D-3/C, Change "A" Service Water Pump to "A" Service Water Header.

**Changed "A" Service Water Pump to "A" Service Water Header where applicable. - RJA (12/17/09)**

Cliff C. - Change stem to make surveillance be a regular "timed" testing of component, as opposed to a "post-repair retest", to ensure examinee does not conclude a new "base line" for data is being established.

**Modified stem per reviewer comments. - RLC 2/22/09**

**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.

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

Question #: **15**

Question ID: **9000010**

RO  **SRO**

Student Handout?

Lower Order?

I-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.

Level of Use  
**Continuous**

STOP THINK ACT REVIEW

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- \_\_\_\_\_ 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

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

Question #: 15

Question ID: 9000010

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 15

Rev. 0

Selected for Exam

Origin: New

Past NRC Exam?

## 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.
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.

7  
3

Level of Use  
**Continuous**

STOP

THINK

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REVIEW

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

Question #: 15

Question ID: 9000010

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 15

Rev. 0

Selected for Exam

Origin: **New**

Past NRC Exam?

## Attachment 1

### Actions for IST Data Outside "Acceptable" Limits

(Sheet 1 of 1)

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

3

5

Level of Use  
**Continuous**

STOP

THINK

ACT

REVIEW

SP 2612A

Rev. 010-08

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

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.

Bruce F. - D-3/C; No comment

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

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

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.

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

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 determine 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.

Bruce F. - D-2/C, No comment

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

Generic K/A Selected

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

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?

**NRC K/A Generic**

System **2.4**

Emergency Procedures /Plan

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  $\Delta T$  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:

- $\Delta T$  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

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REVIEW

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

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?

## Millstone Unit 2 Emergency Generation Reduction

AOP 2557

Revision 006-06

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### 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).

\_\_\_ 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

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

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.

Bruce F. - D-4/C, 50/50 (A and B)

**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

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

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:

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

**ZONE 45**

**WEST DC SWGR  
ROOM FLP-6**

## 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.

Level of Use  
Reference

STOP

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

Question #: 18

Question ID: 9000011

RO

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Lower Order?

I-SRO Ques. # 18

Rev. 0

Selected for Exam

Origin: New

Past NRC Exam?

**ZONE 45**



**WARNING**



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.

Level of Use  
Reference

STOP

THINK

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

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?

**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"

Level of Use  
Reference

STOP

THINK

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

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		Smoke	
	Total No. of Channels	Minimum Channels Operable	Total No. of Channels	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

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

Question #: **19**

Question ID: **9000013**

RO  SRO

Student Handout?

Lower Order?

I-SRO Ques. # **19**

Rev. **1**

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%, then transition to OP 2208, Attachment 5, Reactivity Thumbrules, to continue the power reduction to 55% to secure the Feed Pump.
- 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 to 55% to secure the Feed Pump.
- C** Use Reactivity Plan RE-G-11 to reduce power to 55%, 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 55%, 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.

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

NRC - (original question comments) Modify all choices to suggest use of the 55% Reactivity plan.

RLC - Modified all four choices to allow for "55%" to be added to each, per NRC comments. [11/30/09]

Bruce F. - D-2/C, No comment

**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**    2.1.43        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.

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

Question #: **19**

Question ID: **9000013**

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # **19**

Rev. **1**

Selected for Exam

Origin: **New**

Past NRC Exam?

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

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

Question #: 19

Question ID: 9000013

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 19

Rev. 1

Selected for Exam

Origin: New

Past NRC Exam?

## Millstone Unit 2 Rapid Downpower

AOP 2575

Revision 004-01

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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 preferred, 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.

Level of Use  
Continuous

STOP

THINK

ACT

REVIEW

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

Question #: **19**

Question ID: **9000013**

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # **19**

Rev. **1**

Selected for Exam

Origin: **New**

Past NRC Exam?

## Millstone Unit 2 Rapid Downpower

AOP 2575

Revision 004-01

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### 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 initial 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.
- e. CLOSE CH-501, VCT outlet isolation.
- f. CHECK charging flow at desired rate.
- g. Go To step 3.9.

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

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

\_\_\_ 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.

Bruce F. - D-2/W, 50/50 (C and D)

**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.

# 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?

## 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
    - 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."

③

Level of Use  
Reference

STOP

THINK

ACT

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OP 2209A

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# 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?

## 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.

Level of Use  
Reference

STOP

THINK

ACT

REVIEW

OP 2209A

Rev. 026-06

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

Question #: 21

Question ID: 900024

 RO SRO

Student Handout?

Lower Order?

I-SRO Ques. # 21

Rev. 1

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 In anticipation of a plant trip, ensure "E" & "F" IACs 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

NRC - (original question comments) Add "In anticipation of a plant trip, " to the beginning of Choice "D" and reword to make length comparable.

RLC - Reworded Choice "D" per NRC comments. [11/30/09]

Bruce F. - D-3/C, No comment

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.

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

Question #: 21

Question ID: 9000024

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 21

Rev. 1

Selected for Exam

Origin: New

Past NRC Exam?

**Millstone Unit 2  
Degraded Voltage**

ΔOP 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

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

Question #: 22

Question ID: 53551

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 22

Rev. 2

Selected for Exam

Origin: Bank

Past NRC Exam?

A plant startup from a mid-cycle maintenance outage is in progress and the reactor has just been declared critical. Shortly after going critical, the SPO inadvertently overfeeds the S/Gs, and then regains control. This action results in average RCS temperature lowering to 519 °F and then stabilizing.

Which of the following operator actions should be taken, based on administrative requirements?

- A** Determine RCS Tave is within its limit once per hour until Tave reaches or exceeds 525 °F.
- B** Restore Tavg to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.
- C** Emergency Boration must be commenced immediately.
- D** Immediately trip the Reactor and carry out EOP 2525.

### Justification

A - CORRECT; Per T.S. 3.1.1.5, when ever the reactor is critical, RCS Tavg must be  $\geq 515^{\circ}\text{F}$  at all times. However, the T.S. Surveillance Requirements state that if Tavg drops below  $525^{\circ}\text{F}$ , then Tavg must be verified to be  $\geq 515^{\circ}\text{F}$  (within its limit) at least once per hour.

B - WRONG; This is the required action if Tavg dropped below  $515^{\circ}\text{F}$ .

Plausible; Examinee may believe the limit for action is  $< 525^{\circ}\text{F}$  as that is the reactor startup procedural limit (OP-2202).

C - WRONG; This is the required action if Tavg dropped below  $500^{\circ}\text{F}$ .

Plausible; Examinee may believe an uncontrolled cooldown below the procedural required limit necessitates immediate action to ensure reactivity control.

D - WRONG; This is the required action if the cooldown was not stopped (temperature control is not regained) before Tavg drops below  $500^{\circ}\text{F}$ .

Plausible; Examinee may believe an uncontrolled cooldown below the T.S. limit requires immediate action to ensure plant control.

### References

Tech Spec Surveillance 4.1.1.5

### Comments and Question Modification History

NRC - (original question comments) Question is LOD 5, not required knowledge from memory. Replace question with one testing Tech. Spec. knowledge.

RLC - Original question #220 replaced by Item #53551, rev. 2.

Bruce F. - D-2/C, No comment

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.

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

Question #: 22

Question ID: 53551

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 22

Rev. 2

Selected for Exam

Origin: Bank

Past NRC Exam?

September 25, 2003

## REACTIVITY CONTROL SYSTEMS

### MINIMUM TEMPERATURE FOR CRITICALITY

#### LIMITING CONDITION FOR OPERATION

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3.1.1.5 The Reactor Coolant System temperature ( $T_{avg}$ ) shall be  $\geq 515^{\circ}\text{F}$  when the reactor is critical.

APPLICABILITY: MODES 1 and 2\*.

ACTION:

With the Reactor Coolant System temperature ( $T_{avg}$ )  $< 515^{\circ}\text{F}$ , restore  $T_{avg}$  to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

#### SURVEILLANCE REQUIREMENTS

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4.1.1.5 The Reactor Coolant System temperature ( $T_{avg}$ ) shall be determined to be  $\geq 515^{\circ}\text{F}$ .

- a. Within 15 minutes prior to making the reactor critical, and
- b. At least once per hour when the reactor is critical and the Reactor Coolant System temperature ( $T_{avg}$ ) is  $< 525^{\circ}\text{F}$ .

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\* With  $K_{eff} \geq 1.0$ .

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

Question #: 23

Question ID: 9000026

 RO SRO Student Handout? Lower Order?

I-SRO Ques. # 23

Rev. 0

 Selected for Exam

Origin: New

 Past NRC Exam?

While preparing for an Aerated Waste Monitor Tank (AWMT) discharge, the PEO reported to the Shift Manager that he was unable to access PIOPS for the initial setup of the Aerated Waste System radiation monitor (RM-9116) because the PIOPS keypad appears to be failed. I&C reports it will take several hours to repair or replace the keypad.

Which one of the following actions is required in order for the SM to approve the discharging of the AWMT?

- A A SECOND Chemist must FIRST approve the analysis results and flow requirements stated on the existing AWMT discharge permit.
- B The Chemist must be briefed to ensure the AWMT discharge is sampled 15 minutes AFTER the discharge has commenced.
- C The AWMT may be discharged if the Alert and Alarm limits presently set in PIOPS are lower than the limits the existing permit.
- D The PIOPS keypad must FIRST be repaired to allow completion of the initial setup or another AWMT discharge permit is required.

### Justification

D; CORRECT - SP-2617A requires the setup and limits programmed into the discharge radmonitor by PIOPS (Programmable Input/Output Processing System) be verified, up to and including a check of the I&C parameters for calibration and the performance of a "source check". None of these can be done without accessing PIOPS. Therefore, the keypad must be repaired in order to perform the discharge under the existing permit, or a new permit must be generated following the procedure for an inoperable rad. monitor.

A; WRONG - This action does not meet the administrative requirements for discharging with an inoperable radmonitor. Plausible: The examinee may recognize that the discharge may be performed under the existing conditions if the radmonitor is considered to be inoperable. However, the administrative requirements to discharge with an inoperable radmonitor state that a second sample must be drawn and analyzed by a second Chemist, not just reviewed by a second Chemist.

B; WRONG - This will help verify the radmonitor is operating and tracking the discharge radiation levels properly, but not that it is set up correctly to isolate it if limits are exceeded. Plausible: Examinee may believe this will ensure the limits stated in the discharge permit are met, regardless of the radmonitor setup.

C; WRONG - This would meet the radiological requirements of the discharge if the radmonitor was known to be operable and in calibration. These aspects can not be verified without accessing PIOPS. Plausible: Examinee may believe the setup of the AWMT radmonitor is irrelevant as long as the limits known to exist in PIOPS are more conservative than those on the discharge permit.

Question References not yet listed.

### Comments and Question Modification History

New question generated, based on NRC and Exam Validator comments on previous questions. - RLC 12/23/09

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 2.3.11 RO 2.7 SRO 3.2 CFR Link (CFR: 45.9 / 45.10)

Ability to control radiation releases.

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

Question #: 23

Question ID: 900026

RO

SRO

Student Handout?

Lower Order?

I-SRO Ques. # 23

Rev. 0

Selected for Exam

Origin: New

Past NRC Exam?

## 1. PURPOSE

### 1.1 Objective

This procedure provides instructions to ensure during all MODES of operation all requirements are met prior to and during all radioactive discharges from the following:

- Aerated Liquid Radwaste System (AWMT), to satisfy REMODCM IV.C.1 Surveillance Requirement, Table IV.C-2, items 1.b. and 3.b., "Radioactive Liquid Effluent Monitoring Instrumentation."
- Clean Liquid Radwaste System (CWMTs), to satisfy REMODCM IV.C.1 Surveillance Requirement, Table IV.C-2, items 1.a. and 3.a., "Radioactive Liquid Effluent Monitoring Instrumentation."

### 1.2 Discussion

The objective of this procedure is satisfied by performing the following:

- SOURCE CHECK of applicable radiation monitor prior to discharge:
  - IF discharging CWMT, RE-9049
  - IF discharging AWMT, RE-9116
- Checking flow recorders OPERABLE (tracking flow and indicating), prior discharge:
  - IF discharging CWMT, "FR-9050, SYSTEM DISCHARGE FLOW RECORDER"
  - IF discharging AWMT, "SYSTEM DISCHARGE FLOW, FR-9118"
- CHANNEL CHECK of applicable radiation monitor, 15 minutes after initiation of discharge
- If discharge is performed with radiation monitor *not* OPERABLE, ensuring 2 independent samples have been taken and analyzed
- Performing dual *and* independent verification of necessary steps for all discharges.

CHANNEL CHECK, as used in this procedure, refers to the qualitative assessment of applicable radiation monitor behavior during operation, as compared to other similar indications of the same instrumentation.

The Aerated and Clean Waste Radiation Monitors have design ranges of 10 to  $10^6$  cpm [FSAR Table 7.5-5]. These radiation monitors and PIOPS are *not* OPERABLE above  $10^6$  cpm.

Level of Use  
Continuous

STOP

THINK

ACT

REVIEW

SP 2617A

Rev. 029-06

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

Question #: 24

Question ID: 9000016

RO  SRO

Student Handout?

Lower Order?

I-SRO Ques. # 24

Rev. 2

Selected for Exam

Origin: New

Past NRC Exam?

The plant was operating normally at 100% power with RCS coolant activity at the maximum allowed limit due to known fuel defects. The crew manually tripped the plant due to rapidly lowering Pressurizer pressure and level. EOP 2525, Standard post Trip Actions, was successfully performed and the crew has entered EOP 2534, Steam Generator Tube Rupture.

Why does this procedure require the crew to perform a cooldown on both Steam Generators until both hot leg temperatures are less than or equal to 515°F prior to isolating the affected Steam Generator?

- .....
- A The cooldown can be performed quicker with both S/G allowing earlier isolation of the affected S/G.
  - B If the affected S/G is isolated prior to a cooldown, then increasing radiation levels in the affected S/G will result in a higher release later.
  - C The initial cooldown will significantly lower the affected S/G level resulting in more available volume for the coolant from the tube rupture.
  - D To limit the radiation release due to the potential for lifting the Main Steam Safeties on the affected S/G.

## Justification

D IS CORRECT; The Technical Basis for this step simply states that this is the reason for doing the cooldown to 515°F before isolating the SG.

A is incorrect; While the cooldown can be performed more quickly on two steam Generators versus one Steam Generator, this is NOT the reason for the cooldown to 515°F.

Plausible because this is a true statement; however, it answers the question, "Why use two Steam generators to cool down?" It does NOT answer the question, "Why cool down to 515°F?"

B is incorrect; If the affected Steam Generator were isolated prior to the cooldown, radiation levels would actually be lower later. The higher Steam Generator pressure would limit the leakage of coolant from the RCS to the Steam Generator, resulting in lower activity levels in the affected Steam Generator.

Plausible because the examinee could think that activity levels in the affected Steam Generator would continue to rise resulting in higher activity levels on a later release.

C is incorrect; A potentially true statement, but it is NOT the reason for cooling down to 515°F with both Steam Generators.

Plausible because cooling down the affected Steam Generator could result in less inventory and allow more room for leakage from the RCS.

## References

EOP 2534, Steam Generator Tube Rupture and associated Technical Guide

## Comments and Question Modification History

Potential additional distracter: "The initial cooldown of both Steam Generators ensures that both Steam Generators remain coupled during the subsequent cooldown."

Bruce F. - D-2/C, No comment

**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 2.3.14 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.

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

Question #: 24

Question ID: 9000016

 RO SRO

Student Handout?

Lower Order?

I-SRO Ques. # 24

Rev. 2

Selected for Exam

Origin: New

Past NRC Exam?

## Attachment 5

### EOP 2534, "Steam Generator Tube Rupture," Implementation Guide

(Sheet 1 of 5)

#### Overview/Strategy

1. The primary Safety Functions challenged by a SGTR are RCS inventory control and Containment Isolation. Containment Isolation is re-established by the isolation of the affected SG and reducing RCS pressure below the SG relief valve setpoints.
2. During a NC cooldown with an isolated SG, an inverted  $\Delta T$  ( $T_C$  greater than  $T_H$ ) will be observed due to a small amount of reverse heat transfer in the isolated SG. This has no effect on the NC flow in the intact SG.
3. SGTR Strategy
  - Cooldown RCS to  $< 515^\circ\text{F } T_H$
  - Identify, isolate, and confirm most affected SG
  - Depressurize RCS within 50 psi of isolated SG while maintaining:
    - RCP NPSH limits
    - At least  $30^\circ\text{F}$  subcooling
  - Cooldown to SDC entry conditions

#### Critical Tasks/Operator Credited Actions

1. The most affected SG is required to be isolated within 60 minutes from event initiation. (The design basis assumes a SGTR occurs while operating at full power. Therefore, from time of the reactor trip).
2. The operator is assumed to lower RCS pressure to minimize the primary-secondary leak rate and prevent overfill of the affected SG.
3. The operator cools the RCS using the intact SG and places SDC in service approximately 16 hours following the SGTR.

Level of Use  
Information

STOP

THINK

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## 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

Bruce F. - D-3/C, Need to analyze each answer.

**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

# 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?

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

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### 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.

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

Question #: **25**  
I-SRO Ques. # **25**

Question ID: **9800061**     RO     SRO     Student Handout?     Lower Order?  
Rev. **0**     Selected for Exam    Origin: **Mod**     Past NRC Exam?

## Millstone Unit 2 Diagnostic for Loss of Electrical Power

AOP 2501      Revision 001 – 02  
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### Attachment 1 Lost Control Power (Sheet 1 of 1)

	Loss of VR-11	Loss of VR-21	Loss of VA-10	Loss of VA-20
Diagnostic Indication (deenergized)	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-21 loss.		
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 spnt @ 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 ctrl)	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 1A]
		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 valve 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 control 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 (S/JAE 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