Dominion

April 15, 2009
OT2-09-012

Mr. John Caruso
Lead Examiner
U.S. Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia, PA 19406-1415

Subject: Withdrawal of Comments Regarding Question #35

Dear Mr. Caruso,

This letter is to formally notify you of Millstone's withdrawal of comments regarding question #35 on the 2009 LOIT exam administered at Millstone Power Station on February 24, 2009.

The Millstone training staff will continue to investigate any technical issues associated with the question and, as necessary, modify training materials or initiate necessary corrective actions if appropriate.

Please do not hesitate to contact me if you need any additional information.

Sincerely,

Jeff T. Spence
Manager, Nuclear Training
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385
(860) 437-2540

**Attachment 5**

**Post Examination Comments and Supporting Documentation**

**Millstone Power Station Unit 2**
**Dominion Nuclear Connecticut, Inc. (DNC)**

Final Submittal for SRO Question #77 (#2): Rev 1 Revised 3/31/09

While performing a plant cooldown, the crew was about to place Shutdown Cooling (SDC) in service with "A" and "B" Reactor Coolant Pumps (RCPs) in service, when Pressurizer Pressure Narrow Range Instrument, P-103-1, suddenly fails high.

Which of the following describes the impact of this failure and the action that the US must direct to stabilize the plant?

☑ **A** The "A" Power Operated Relief Valve (PORV) will open and RCS pressure will rapidly lower. Place the "A" LTOP switch to "High", close PORV Block Valve, secure the "A" and "B" RCPs, and place SDC in service.

☐ **B** The Facility 1 Safety Injection Tank (SIT) outlet Valves will open causing the SITs to inject. Override and close the SIT Outlet Valves and restore Pressurizer level to the desired setpoint.

☐ **C** SDC Suction Isolation Valve, SI-652, will NOT open until the high pressure input is cleared. Direct I&C to bypass or disconnect Pressurizer Pressure Narrow Range Instrument, P-103-1.

☐ **D** The "A" Power Operated Relief Valve (PORV) will open and RCS pressure will rapidly lower. Immediately place the associated PORV Block Valve to 'Close' and log into the LTOP Technical Specification LCO.

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Submitted Answer Justification:

A is correct. In this alignment, the LTOP switch is in the LOW position, resulting in the low pressure setpoint being applied to the PORVs. If Pressurizer Pressure Narrow Range Instrument, P-103-1, suddenly fails high, the "A" PORV will open resulting in RCS pressure rapidly lowering. The PORV can only be closed by placing the "A" LTOP switch in the HIGH position. The rapid reduction in RCS pressure will result in the loss of NPSH for the RCPs, so they must be secured. With the loss of the RCPs, core cooling must be established by initiating SDC.

B is incorrect. Although the Facility 1 SIT Outlet Valves get an open signal from the failed pressure instrument, the present plant alignment requires the pressure input to be bypassed to prevent inadvertent opening.

C is incorrect. The high pressure input to the SDC Suction Isolation Valves has been removed. Additionally, P-103-1 (Facility 1) does NOT align with SI-652 (Facility 2).

D is incorrect. The "B" PORV will NOT open as it is NOT associated with P-103-1. Additionally, if P-103 failed high, attempting to close the "B" PORV would NOT work because it is interlocked open with the "B" LTOP switch set to LOW.

**Licensee's Justification For Change For SRO Question #77 (#2):**

The original correct answer for this question is choice "A". The original draft answer did not include the phrase, "*and place SDC in service.*" In-house validation suggested that this phrase be added to complete 'the action that the US must direct to stabilize the plant'; however, SDC cannot be placed in service with Pressurizer Pressure instrument, P-103-1, failed high. As a result of the instrument failure, SDC Isolation Valve, SI-651, will be interlocked closed, preventing SDC from being placed in service. This error was not recognized during subsequent validation.

The stem of the question asks what must be done "*to stabilize the plant.*" While the actions to "*place SDC in service*" would certainly stabilize the plant, the time required to do so to meet choice "A" are unrealistic. Manual operation of Shutdown Cooling Isolation Valve, SI-652 in Containment would far exceed allowable radiological dose (200 - 300 mr/hr) and the time necessary to determine inputs to the valve from P-103-1 would require in-depth circuit analysis since there are no procedures readily available to override or defeat this interlock. 1

The question does not indicate what RCS temperature is at the time of the instrument failure. Therefore, in this situation, once RCPs are secured it can be assumed that the cooldown would be stopped and RCS temperature stabilized using the steam generators while natural circulation develops. This was not a selection option for this question.

If Shutdown Cooling Isolation Valve, SI-651, cannot be opened due to the failure of Pressurizer Pressure instrument, P-103-1, then, obviously, Shutdown Cooling cannot be placed in service to meet choice "A"; therefore, choice "A" is incorrect. All other answers have been validated as being incorrect; therefore; SRO question #77 (2) has NO correct answer.

Additional documentation is attached.

**The following highlighted excerpts from the Shutdown Cooling lesson plan, SDC-00-C, demonstrate that Shutdown Cooling Isolation Valve, SI-651, cannot be opened with Pressurizer Pressure Instrument, P-103-1, failed above 280 psia.**

| **d. Control and Instruments** | |
|---|---|
| The Suction Line Isolation Valve motor operators are powered from MCC B61 (SI-652) and MCC B51 (SI-651).<br><br>The 480V power supply to 2-SI-652 is normally isolated in order to prevent inadvertent opening and subsequent system over pressurization during power operation. This isolation switch is located in the Control Room behind panel C80. Operation of this switch will actuate annunciator D-39 on C-01 whenever power is made available to 2-SI-652.<br><br>To address single-failure concerns for post-accident boron precipitation, 2-SI-651 is equipped with an alternate power supply from 480 volt MCC B61. An administratively controlled kirk-key shifts control to local panel and bypasses the RCS pressure interlock (280 psia). | PEO-4B |
| The control switches for SI-651 and SI-652 are located in the Control Room on panel C-01. They are two position OPEN/CLOSE keylock switches. Each of the valves has RED (open) and GREEN (closed) indication lights. | RO-2A |
| Procedurally, the LPSI system, SDC heat exchangers, and interconnecting piping can not be exposed to an RCS pressure greater than 265 psia or temperature greater than 300° F. To protect the piping against excessive pressure SI-651 and SI-652 are interlocked with the low range Pressurizer pressure channels (P-103 and P-103-1). The range of these instruments is 0 to 1600 psia. Pressure is displayed on C-03 and on C-21 and can also be monitored on the Plant Process Computer. The SDC Suction Isolation Valves are interlocked closed if RCS pressure is greater than 280 psia. If SI-651 or SI-652 were already open (i.e. SDC in service) and pressure rose above 280 psia an alarm is received on C01 alerting the operators. | |
| Each Suction Isolation Valve motor operator has a manual handwheel and clutch mechanism allowing local operation of the valve. | |

## Attachment 3 - Interlocks & Automatic Features

| Instrument # | Feature |
|---|---|
| LT-3001<br>LT-3002<br>LT-3003<br>LT-3004 | Two out of four Refueling Water Storage Tank (RWST) level indications less than setpoint will initiate a Sump Recirculation Actuation Signal (SRAS) from the Engineered Safeguards Actuation System (ESAS).<br>Provided ESAS were not bypassed and a signal were to occur, the following Shutdown Cooling System components would be affected:<br>• LPSI Pump "A" and "B" would trip;<br>• SI/CS Minimum flow recirc valves (2-SI-659 & 2-SI-660) would close (provided that HS-3659A and HS-3660A are in the "OPER" position);<br>• Containment Sump Outlet Valves (CS-16.1 A/B) open.<br>• Shutdown Cooling (SDC) Heat Exchanger RBCCW outlet valves would open (RB-13.1 A/B). |
| PT-102A<br>PT-102B<br>PT-102C<br>PT-102D | Two out of four Pressurizer Pressure indications less than setpoint will initiate SIAS from the Engineered Safeguards Actuation System (ESAS). SIAS is blocked during SDC operation, but if it was not blocked, the following components within the LPSI system or affecting LPSI system performance would be effected:<br>• LPSI Pump "A" and "B" start.<br>• LPSI injection valves (2-SI-615, 2-SI-625, 2-SI-635, & 2-SI-645) open.<br>• ESF Room Fans (F-15A & F-15B) start. |
| PT-103 | Prevents the shutdown cooling suction line isolation valve 2-SI-652 from being opened if a high Reactor Coolant System (RCS) pressure condition exists (setpoint: 280 psia increasing). |
| PT-103-1 | Prevents the shutdown cooling suction line isolation valve 2-SI-651 from being opened if a high Reactor Coolant System (RCS) pressure condition exists (setpoint: 280 psia increasing). |

Final Submittal for SRO Question #98 (#23):

A Fuel Handling Accident has occurred inside Containment as evidenced by several Containment process and area radiation monitors rising or in alarm. All personnel were immediately evacuated.

In order to limit personnel exposure, per AOP-2577 (Fuel Handling Accident), the Shift Manager must ensure which of the following are performed? 1

☑ **A** The Containment (Outage) Equipment Hatch must be closed no more than 30 minutes after the event.

☐ **B** Containment Purge valves, AC-4, 5, 6, and 7 must be closed no more than 50 minutes after the event.

☐ **C** The Transfer Tube Isolation Valve, RW-280, must be closed no more than 50 minutes after the Transfer Carriage is in the SFP

☐ **D** One train of CRACS must be operating in Recirculation mode no more than 30 minutes after the event.

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Submitted Answer Justification:

A is correct. Per AOP-2577 (Fuel Handling Accident), Rev. 008-02, Section 1.0, Containment must be isolated within 30 min. of the event.

B is incorrect. The Containment Purge Valves would be closed for a Fuel Handling Accident in Containment, but they would likely have closed automatically. However, if the auto actuation has not occurred, they must be manually closed immediately.

C is incorrect. The Transfer Tube Isolation Valve, RW-280, is not included in the components that must be closed for Containment Isolation in a Fuel Handling Accident as the opening is low in the refuel pool. This valve would be closed for a loss of Refuel Pool level and takes about 30 minutes to close.

D is incorrect. The calculation for the Control Room radiological exposure following a Fuel Handling Accident is based on having at least one Control Room Air Conditioning train operating in the Recirculation mode within 60 minutes of the event. The calculations used to require recirc mode within 10 minutes. However, new calculations recently changed the requirement to 50 minutes with a 10 minute penalty for other system alignment complications. (AOP-2577 had NOT yet been updated when the question was written.)

**Licensee's Justification For Change For SRO Question #98 (#23):**

The original correct answer for this question is choice "A"; however, given that the stem of the question asks the examinee what must be done "*to limit personnel exposure*", it could be argued that choice "D" is also correct. Choice "D" states, "*One train of CRAC must be operating in Recirculation mode no more than 30 minutes after the event.*" This is not an incorrect action in order to limit personnel exposure.

In 2007, Millstone Unit 2 (MPS2) implemented License Amendment No. 298 to Technical Specifications (TAC NO. 2346). This amendment revised the MPS2 licensing basis in the area of radiological dose analysis for design-basis accidents using the alternative source term (AST) as permitted by 10CFR50.67.

Full implementation of the AST at MPS2, as defined in Section 1.2.1 of RG 1.183, required re-analysis of the following accidents: LOCA, Fuel Handling Accident (FHA), Cask Tip Accident, SGTR, MSLB, CREA. In all of the re-analysis performed, for accidents which required the Control Room Air Conditioning System (CRACS) to be placed in the filtered, recirculation mode, the required time for operator action was changed from the previous 10 minute requirement to "within 1 hour".

AOP 2577, "Fuel Handling Accident," includes as part of the 'Discussion' section the following statement:

> "*Control Room radiological exposure following a fuel handling accident in containment is calculated on having at least one train of CRACS operating in the recirculation mode within 10 minutes of the accident.*"

This statement within AOP 2577 is only a statement of fact from the previous requirement before implementation of the AST approach at MPS2. Procedure step 3.5 in AOP 2577 includes actions necessary to place the CRAC System in filtered recirculation; however, there is no specified time requirement for implementation of this step.

Prior to attending license school, the SRO Upgrade candidates attended a presentation (C07601), as part of Licensed Operator Requalification Training concerning the Alternate Source Term and the affect on Control Room Air Conditioning (CRAC) System and the time change to "within 1 hour" for filtered recirculation.

Additionally, OP 2260, "EOP User's Guide" was revised to reflect the re-analysis and implementation of the AST. The time chosen was a conservative 50 minutes (rather than 1 hour by the accident analysis), to place the CRAC System in filtered recirculation for LOCA, SGTR, and MSLB accidents.

Since placing the CRAC System in filtered recirculation mode is credited as part of the accident analysis for FHA to limit personnel exposure, and given the absence of any procedure directed time requirement within AOP 2577, it can be argued that choice "D" is also an acceptable answer, since 30 minutes is "within 1 hour" as outlined in the accident analysis. Therefore, examinees should be given credit for both choice "A" and choice "D".

A post-exam Condition Report (CR325747) has been generated to identify the modification required for AOP 2577, "Fuel Handling Accident."

Additional documentation is attached.

**The following excerpt is from Technical Specification Bases 3.7.6:**

PLANT SYSTEMS

BASES

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room. For all postulated design basis accidents, the radiation exposure to personnel occupying the control room shall be 5 rem TEDE or less consistent with the requirements of 10 CFR 50.67

The Control Room Envelope (CRE) is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and other non-critical areas including adjacent support offices, and utility rooms. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, ceiling, ducting, valves, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

In order for the control room emergency ventilation systems to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

TS LCO 3.7.6.1 is modified by a footnote allowing the CRE boundary to be opened intermittently under administrative controls. This footnote only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

ACTIONS a., b., and c. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. ACTIONS d. and e. are applicable in MODES 5 and 6, or when recently irradiated fuel assemblies are being moved. The control room emergency ventilation system is required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis), due to radioactive decay. The term "recently" is defined as all irradiated fuel assemblies, until analysis is performed to determine

a specific time. The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm ± 10%.

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), will require operator action to return to a full train lineup. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train lineup. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until 1 hour into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day allowed outage time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day allowed outage time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary. Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

Surveillance Requirement 4.7.6.1.c.1 dictates the test frequency, methods and acceptance criteria for the Control Room Emergency Ventilation System trains (cleanup trains). These criteria all originate in the Regulatory Position sections of Regulatory Guide 1.52, Rev. 2, March 1978 as discussed below.

Section C.5.a requires a visual inspection of the cleanup system be made before the following tests, in accordance with the provisions of section 5 of ANSI N510-1975:

- in-place air flow distribution test
- DOP test
- activated carbon adsorber section leak test

**The following excerpt is from LBDCR #04-MP2-011, AR #07003671-04:**



dose will be retained with the acceptance criteria remaining as "substantially below 10 CFR Part 100 guidelines".

The proposed licensing and plant operational changes are discussed in Section 2.0. These changes require appropriate changes to the MPS2 operating license and technical specifications, which are also described in Section 2.0 of this report. The changes considered as a result of the re-analyses and clarifications are listed below:

a. Revise definition of Dose Equivalent I-131 in Section 1.19 of the Technical Specifications Definitions to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988, as the source of inhalation thyroid dose conversion factors (Reference 8).

b. Change the combined leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths in Technical Specification LCO 3.6.1.2, "Containment Leakage" from < 0.0072 $L_a$ to < 0.014 $L_a$.

c. Change the control room unfiltered in-leakage limit in Technical Specification SR 4.7.6.1.e.3 from 130 SCFM to 200 SCFM and clarify the applicability.

d. Revise the primary to secondary leakage limit in Technical Specification LCO 3.4.6.2.c from 0.035 GPM to 75 GPD through any one steam generator.

e. Approve a new operator action to isolate the control room up to 4 hours after a MSLB.

f. Approve a timing change to an existing operator action such that within 1 hour (instead of 10 minutes) of isolation, the control room emergency ventilation system is placed in filtered recirculation.

g. Delete Technical Specification LCO 3.3.4 and SR 3.3.4, "Containment Purge Valve Isolation Signal," since its intent is addressed by LCO 3.6.3.2, "Containment Ventilation System."

h. Revise the language of Technical Specification LCO 3.3.3.1 Action b to be consistent with similar action statements within that technical specification.

i. Revise the technical specification bases to reflect the above listed changes, as appropriate, in accordance with Technical Specification 6.23, "Bases Control Program."

# Q-77/2 (I-SRO/SRO)

SRO Question #2 I-SRO Question #77

While performing a plant cooldown, the crew was about to place Shutdown Cooling (SDC) in service with "A" and "B" Reactor Coolant Pumps (RCPs) in service when Pressurizer Pressure Narrow Range Instrument, P-103-1, suddenly fails high.

Which of the following describes the impact of this failure and the action that the US must direct to stabilize the plant?

(A) The "A" Power Operated Relief Valve (PORV) will open and RCS pressure will rapidly lower. Place the "A" LTOP switch to 'High', close PORV Block Valve, secure the "A" & "B" RCPs, and place SDC in service.

**Correct Answer as given, which is believed to be invalid**

B The Facility 1 Safety Injection Tank (SIT) Outlet Valves will open causing the SITs to inject. Override and close the SIT Outlet Valves and restore Pressurizer level to the desired setpoint.

C SDC Suction Isolation Valve, SI-652, will NOT open until the high pressure input is cleared. Direct I&C to bypass or disconnect Pressurizer Pressure Narrow Range Instrument, P-103-1.

D The "B" Power Operated Relief Valve (PORV) will open and RCS pressure will rapidly lower. Immediately place the associated PORV Block Valve to 'Close' and log into the LTOP Technical Specification LCO.

# Q-77/2 (I-SRO/SRO)

Slide is from Reactor Coolant System L.P. "RCS-00-C"

## Pressurizer Pressure Instruments

- 2 transmitters (PT-103/103-1) provide signals to:
  - Low temperature over pressure protection.
  - Low range (0-1600 psia) indication on C03.
  - Indication of both on C-21 panel.
  - SDC suction valve interlock
    - Prevent suction valve from opening with RCS >265 psia.
    - Alarm if on SDC and > 280 psia.
  - Auto opening of SIT isolation valves at 300 psia increasing.

# Q-77/2 (I-SRO/SRO)

## DOMINION NUCLEAR CONNECTICUT

**MILLSTONE 2**

**Lesson Title:** Shutdown Cooling System

**Revision:** 4/2

**ID Number:** SDC-00-C

**See *Page 10* of 82 for information on how P-103 and P-103-1 effect the SDC Suction Valves, SI-651 and SI-652**



| | | |
|---|---|---|
| Submitted by: | Pete Strickland | 10/31/07 |
| | Developer | Date |
| Reviewed by: | Sandy Doboe | 11/05/07 |
| | Technical Reviewer | Date |
| Reviewed by: | N/A | N/A |
| | Cognizant Plant Supervisor *(Optional)* | Date |
| Approved by: | Mike Cote | 11/05/07 |
| | Training Supervisor | Date |

The next slide is a page from the Shutdown Cooling System Text, SDC-00-C

Explanation of How P-103 & P-103-1 Effect SI-651 & SI-652

# Q-77/2 (I-SRO/SRO)



Outside containment, there is a 12 inch manual isolation gate valve (2-SI-709) in the LPSI pump suction line. This SDC Suction Header Isolation valve provides for Containment isolation during power operations.

**f. Control and Instruments**

The Suction Line Isolation Valve motor operators are pow
MCC B61 (SI-652) and MCC B51 (SI-651).

The 480V power supply to 2-SI-652 is normally isolated in
prevent inadvertent opening and subsequent system over
pressurization during power operation. This isolation swit
in the Control Room behind panel C80. Operation of this
actuate annunciator D-39 on C-01 whenever power is ma
to 2-SI-652.

To address single-failure concerns for post-accident bo
precipitation, 2-SI-651 is equipped with an alternate pow
from 480 volt MCC B61. An administratively controlled
shifts control to local panel and bypasses the RCS pres
interlock (280 psia).

The control switches for SI-651 and SI-652 are located in the Control Room on panel C-01. They are two position OPEN/CLOSE keylock switches. Each of the valves has RED (open) and GREEN (closed) indication lights.

RO-2A

Procedurally, the LPSI system, SDC heat exchangers, and interconnecting piping can not be exposed to an RCS pressure greater than 265 psia or temperature greater than 300° F. To protect the piping against excessive pressure SI-651 and SI-652 are interlocked with the low range Pressurizer pressure channels (P-103 and P-103-1). The range of these instruments is 0 to 1600 psia. Pressure is displayed on C-03 and on C-21 and can also be monitored on the Plant Process Computer. The SDC Suction Isolation Valves are interlocked closed if RCS pressure is greater than 280 psia. If SI-651 or SI-652 were already open (i.e. SDC in service) and pressure rose above 280 psia an alarm is received on C01 alerting the operators.

Each Suction Isolation Valve motor operator has a manual handwheel and clutch mechanism allowing local operation of the valve.

Effect of P-103 & P-103-1 High Pressure (real or failed) with SI-651 & SI-652 *CLOSED*

greater than 265 psia or temperature greater than 300° F. To protect the piping against excessive pressure SI-651 and SI-652 are interlocked with the low range Pressurizer pressure channels (P-103 and P-103-1). The range of these instruments is 0 to 1600 psia. Pressure is displayed on C-03 and on C-21 and can also be monitored on the Plant Process Computer. The SDC Suction Isolation Valves are interlocked closed if RCS pressure is greater than 280 psia. If SI-651 or SI-652 were already open (i.e. SDC in

Effect of P-103 & P-103-1 High Pressure (real or failed) with SI-651 & SI-652 *OPEN*

than 280 psia. If SI-651 or SI-652 were already open (i.e. SDC in service) and pressure rose above 280 psia an alarm is received on C01 alerting the operators.

# Q-77/2 (I-SRO/SRO)

**Setpoint:**

- Pressurizer pressure greater than 280 psia with 2–SI–651 open
- P103–1 deenergizes

**C–9**

**SI–651 OPEN**

**AUTOMATIC FUNCTIONS**

1. None

**CORRECTIVE ACTIONS**

1. OBSERVE "SDC SYS SUCT SYS ISOL, SI–651," position (C–01).
2. IF "SDC SYS SUCT SYS ISOL, SI–651" is *not* desired to be open, CLOSE "SDC SYS SUCT SYS ISOL, SI–651" (C–01).
3. ENSURE "SDC SYS SUCT SYS ISOL, SI–651," closes and alarm clears (C–01).
4. IF annunciator does *not* clear AND "SDC SYS SUCT SYS ISOL, SI–651," is open, REDUCE RCS pressure to less than 280 psia.
5. IF annunciator does *not* clear AND "SDC SYS SUCT SYS ISOL, SI–651," is closed, SUBMIT Priority 2 Trouble Report to I&C Department.

**SUPPORTING INFORMATION**

1. Initiating Devices
   - 2–SI–651 limit switch and pressurizer pressure, P103–1 bistable, PA–103–1C
   - 63X/P103–1
2. Computer Points
   - SI651 (digital)
3. Procedures
   - OP 2310, "Shutdown Cooling System"
4. Control Room Drawings
   - 25203–26015, Sheet 3
   - 25203–32008, Sheet 33
5. Annunciator Card Location: TB2–J18

ARP 2590A–035
Rev. 000
Page 1 of 1

Alarm response for P-103-1 "high" and with SI-651 *OPEN*. Loss of Shutdown Cooling Procedure, AOP-2572, is not referenced.

SI-651 will not open until the failed instrument, P-103-1, is repaired or bypassed by I&C.

# Q-77/2 (I-SRO/SRO)

**Setpoint:**

- Pressurizer pressure greater than 280 psia with 2–SI–652 open
- P103 deenergizes

**D–9**

**SI–652 OPEN**

**AUTOMATIC FUNCTIONS**

1. None

**CORRECTIVE ACTIONS**

1. OBSERVE "SDC SYS SUCT CTMT ISOL, SI–652," position (C–01).
2. IF "SDC SYS SUCT CTMT ISOL, SI–652" is *not* desired to be open, CLOSE "SDC SYS SUCT CTMT ISOL, SI–652" (C–01).
3. ENSURE "SDC SYS SUCT CTMT ISOL, SI–652," closes and alarm clears (C–01).
4. IF annunciator does *not* clear AND "SDC SYS SUCT CTMT ISOL, SI–652," is open, REDUCE RCS pressure to less than 280 psia.
5. IF annunciator does *not* clear AND "SDC SYS SUCT CTMT ISOL, SI–652," is closed, SUBMIT Priority 2 Trouble Report to I&C Department.

**SUPPORTING INFORMATION**

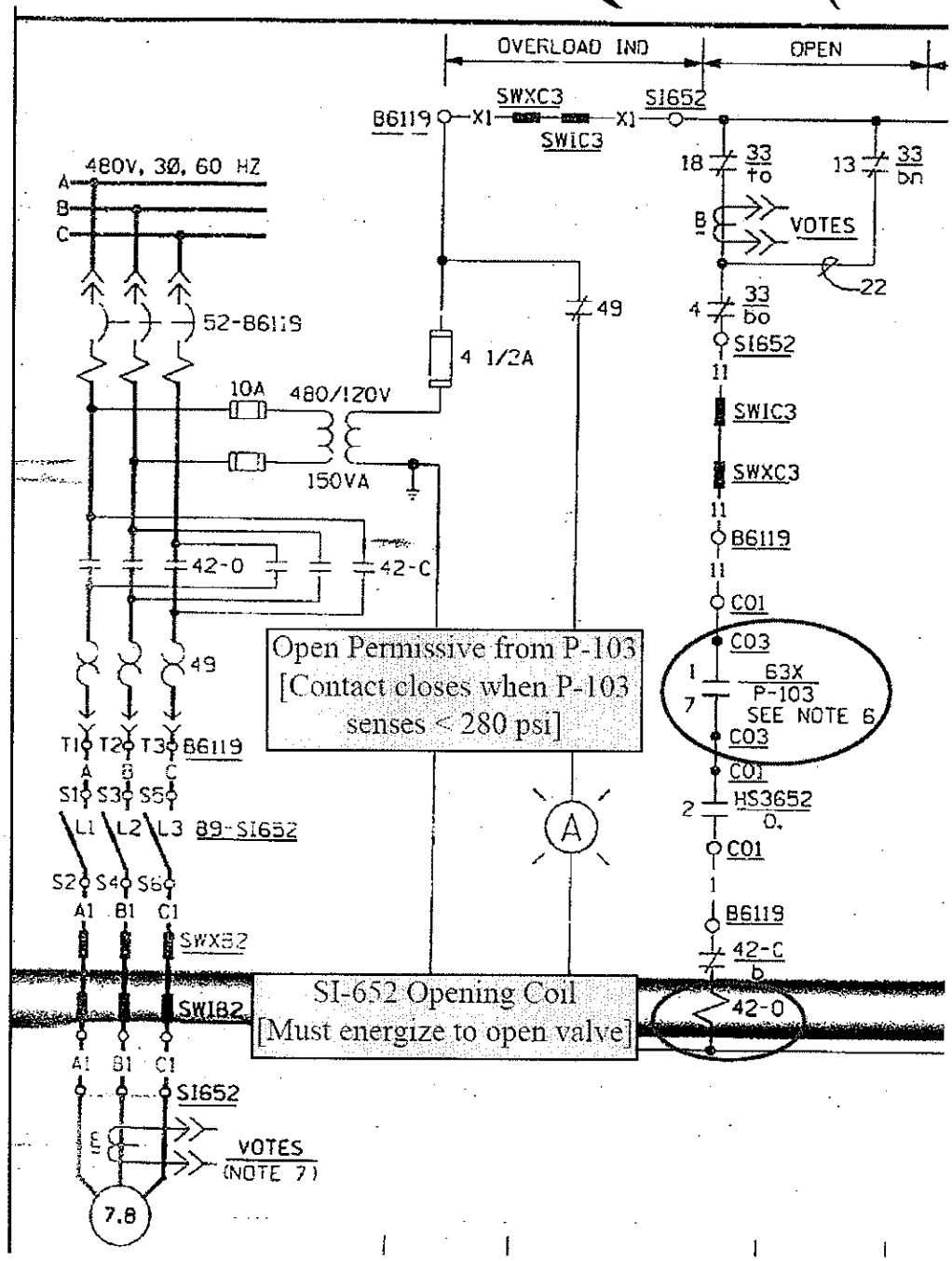
1. Initiating Devices
   - 2–SI–652 limit switch and pressurizer pressure P103, bistable, PA–103–C
   - 63X/P103
2. Computer Points
   - SI652 (digital)
3. Procedures
   - OP 2310, "Shutdown Cooling System"
4. Control Room Drawings
   - 25203–26015, Sheet 3
   - 25203–32008, Sheet 33 → **Next Slide**
5. Annunciator Card Location: TB2–J18

ARP 2590A–036
Rev. 000
Page 1 of 1

Alarm response for P-103 "high" and with SI-652 *OPEN*. Loss of Shutdown Cooling Procedure, AOP-2572, is not referenced.

SI-652 will not open until the failed instrument, P-103, is repaired or bypassed by I&C.

# Q-77/2 (I-SRO/SRO)




Cut-Away of P-103 RCS Pressure input to SI-652 *OPEN* permissive, from P&ID 25203-32008, Sh. 33

SI-652 can not be opened until the failed instrument, P-103, is repaired or bypassed by I&C.

# Q-77/2 (I-SRO/SRO)

MILLSTONE POWER STATION
ABNORMAL OPERATING PROCEDURE

**Loss of Shutdown Cooling**

**AOP 2572**
**Rev. 009 – 03**

Approval Date: 03/27/08

Effective Date: 06/18/08

Level of Use
**C**ontinuous

The next slide is a page from the Loss of Shutdown Cooling Abnormal Operating Procedure, AOP-2572

This is the procedure the operators would normally use for guidance on the recovery of SDC as an RCS heat sink.

# Q-77/2 (I-SRO/SRO)



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Because there is no impact on the Shutdown Cooling System if P-103 or P-103-1 were to "fail high" with SI-651 & SI-652 *OPEN*, the Loss of Shutdown Cooling Procedure, AOP-2572, does not have guidance for this failure.

With SI-651 or SI-652 closed, SDC would not yet be in service. This scenario is addressed by the Alarm Response Procedure (following slide).

The question did not intend to solicit operator actions outside of written procedural guidance.

# Q-98/23 (I-SRO/SRO)

SRO Question #23 I-SRO Question #98

A Fuel Handling Accident has occurred inside Containment as evidenced by several Containment process and area radiation monitors rising or in alarm. All personnel were immediately evacuated.

In order to limit personnel exposure, per AOP-2577 (Fuel Handling Accident), the Shift Manager must ensure which of the following are performed?

(A) The Containment (Outage) Equipment Hatch must be closed no more than 30 minutes after the event. **Correct Answer as given**

B Containment Purge Valves, AC-4, 5, 6, and 7 must be closed no more than 50 minutes after the event.

C The Transfer Tube Isolation Valve, RW-280, must be closed no more than 50 minutes after the event.

(D) One train of CRACS must be operating in Recirculation mode no more than 30 minutes after the event. **Suggested Additional Correct Answer**

# Q-98/23 (I-SRO/SRO)

The next three slides are excerpts from the "Alternate Source Term" presentation given in License Operator Requalification Training (LORT) that all License Upgrade candidates attended.

# Tech. Spec. Changes –
## "Alternate Source Term"

## Cycle 07-6 Operator Training
## C07601

# Q-98/23 (I-SRO/SRO)
## Tech. Spec. Changes –
### "Alternate Source Term"

'Partial implementation' overview

- Done in Sept. – Nov. 2004
  - Initial changes to allow relaxation of CTMT and SFP closure during outages.
    - Change calculated doses due to FHA in CTMT or SFP
    - Change times for isolating and going to recirc. in CR
    - Changes due to a cask drop in SFP

# Q-98/23 (I-SRO/SRO)
# Tech. Spec. Changes –
## "Alternate Source Term"

### 'Full implementation phase' overview

- Done 08/29/07
  - Implement Reg. Guide 1.183 as basis for source term.
  - Allow a consistent Design Basis for all accident dose assessments.
  - Allow more Cont. Room In-leakage
  - Allow more CTMT by-pass leakage
  - Allow more RWST valve backleakage (IST)
  - Increased Pri.-to-Sec. Leakrate limit
  - Re-define Dose Equivalent I-131
  - Allow more time, (1 hr), for Cont. Rm. Isolation & recirc.
  - Allow more time for termination of CS (4 hrs)

This is the change that effected question #98/23

# Q-98/23 (I-SRO/SRO)

# Tech. Spec. Changes –

## "Alternate Source Term"

## Technical Specification Basis Changes

~~LBDCR 04-MP2-016~~
~~February 24, 2005~~

PLANT SYSTEMS

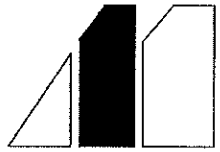
BASES

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), operator action will be required to return to a full train lineup. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train lineup. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until ~~10 minutes~~ 1 hour into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

MILLSTONE POWER STATION
GENERAL OPERATING PROCEDURE



See pages 20 and 21 for changes made to this procedure based on the change to the Alternate Source Term

**Unit 2 EOP User's Guide**

**OP 2260**

**Rev. 009−03**

The objective of this procedure is to provide guidance to the operator for Millstone Unit 2 Emergency Operating Procedure (EOP) usage.

STOP THINK ACT REVIEW

Approval Date: 10/28/08

Effective Date: 10/30/08

Level of Use
Information

# Q-98/23 (I-SRO/SRO)

The next slide is page 21 from the Unit 2 EOP User's Guide Procedure, OP-2260

OP-2260 contains information that explains what changed to alter the correct choice for Question #98.

# Q-98/23 (I-SRO/SRO)

The is a page from the Unit 2 EOP User's Guide Procedure, OP-2260

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

**Overview/Strategy**

This procedure contains a check of Safety Functions against acceptance criteria to give the operator a complete status regarding plant conditions and safety. The acceptance criteria are chosen to be accessible from the Control Room and to require no interpretation or interpolation by the operator. The steps that monitor and maintain Safety Functions are called Immediate Actions. These steps are continuously applicable while in SPTA. This procedure also contains Contingency Actions that can be performed to restore those Safety Functions in jeopardy.

The check of Safety Functions provides the operator with objective decision criteria to determine if action is required in the short term to restore plant safety. This permits crisp, reliable decision–making and precludes unnecessary operator action.

The check of Safety Functions discriminates between an uncomplicated reactor trip and more complex events. The Safety Function acceptance criteria are chosen to be consistent with the plant conditions that prevail only in the short term after an uncomplicated reactor trip. Thus, if equipment failures require attention, the acceptance criteria may not be satisfied. The US should determine if the equipment failure is the reason the Safety Function is not met or an actual event is in progress.

**Critical Tasks/Operator Credited Actions**

1. The operator is required to trip all RCPs within 5 minutes of the loss of NPSH, following a Small Break LOCA (SBLOCA).

2. 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. (1)

3. 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.

4. Control Room Air Conditioning System (CRACS) must be initiated (normal or recirculation mode) within 25 minutes of a loss of control room cooling to ensure control room temperature limits are not exceeded. CRAC must be initiated in it's recirculation mode within 50 minutes of receipt of a SIAS signal (due to LOCA, SGTR, or MSLB), and within 4 hours of a MSLB outside of CTMT (if a SIAS has *not* occurred). (3) (2)

Change #3 was made to accommodate the change in the Alternate Source Term.

# Q-98/23 (I-SRO/SRO)

The next three slides are excerpts from the Fuel Handling Accident Abnormal Operating Procedure, AOP-2577

AOP-2577 contains information used to develop the original question. The following slides also explain what changed to alter the correct choice.

MILLSTONE POWER STATION
ABNORMAL OPERATING PROCEDURE

Please see comments on pages 3, 6 and 7.

**Fuel Handling Accident**

**AOP 2577**
**Rev. 008–02**

STOP THINK ACT REVIEW

Approval Date: 07/23/07

Effective Date: 08/29/07

Level of Use
**C**ontinuous

# Q-98/23 (I-SRO/SRO)

## Excerpts from the Fuel Handling Accident Abnormal Operating Procedure, AOP-2577

| Millstone Unit 2<br>Fuel Handling Accident | AOP 2577 Revision 008−02<br>Page 3 of 15 |
|---|---|

This is what the original question was based on.

**1.0 PURPOSE**

1.1 **Objective**

This procedure provides instructions for a fuel handling accident in containment or the spent fuel pool. The isolation of containment or spent fuel pool, will occur within 30 minutes following a fuel handling accident.

1.2 **Discussion**

The design basis fuel handling accident occurs 72 hours following shutdown and results in the failure of all fuel rods.

Control Room radiological exposure following a fuel handling accident in containment is calculated on having at least one train of CRACS operating in the recirculation mode within 10 minutes of the accident.

During the lecture on "Alternate Source Term" (C07601) the Examinees were told this had changed to within 1 hour.

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 or the outage equipment door is installed.
- 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 an isolation valve, blind flange, manual valve, or special device, or
  - Capable of being closed by an operable automatic containment purge valve.

1.3 **Applicability**

This procedure is applicable any time fuel is being moved within the spent fuel pool or containment.

Level of Use
**C**ontinuous

STOP THINK ACT REVIEW

Information used to develop the original question.

Information that changed in Administrative Requirements, but has <u>not</u> yet been changed in AOP-2577.

# Q-98/23 (I-SRO/SRO)

## Excerpts from the Fuel Handling Accident Abnormal Operating Procedure, AOP-2577

| Millstone Unit 2<br>Fuel Handling Accident | AOP 2577 Revision 008−02<br>Page 6 of 15 |
|---|---|

| INSTRUCTIONS | CONTINGENCY ACTIONS |
|---|---|
| ___3.5 VERIFY at least *one* train of CRACS operating in recirculation mode. (C25) | |

**Facility 1**

- HV−203A, Fan F−21A exhaust damper is open.
- Fan F−21A, supply fan is running.
- HV−206A, Fan F−31A exhaust damper is open.
- Fan F−31A, exhaust fan is running.
- HV−212A, Fan F−32A exhaust damper is open.
- Fan F−32A, filter fan is running.
- HV−202, minimum fresh air damper is closed.
- HV−207, cable vault exhaust damper is closed.
- HV−208, exhaust air damper is closed.

(continue)

**Step 3.5, which instructs the operators to place the CRACS in "Recirculation mode", is before Step 3.6, which instructs the operators to verify "Containment Closure"**

Step 3.6 on next page

Level of Use
**C**ontinuous

STOP THINK ACT REVIEW

Procedure use guidelines dictate that, due to the order of the steps in the procedure, the action to place CRACS in recirculation mode must be started before the action to close the Containment Equipment Hatch (VERIFY Containment Closure established).

Additional procedural guidance given in Step 3.6, next slide.

# Q-98/23 (I-SRO/SRO)

## Excerpts from the Fuel Handling Accident Abnormal Operating Procedure, AOP-2577

| Millstone Unit 2<br>Fuel Handling Accident | AOP 2577 Revision 008–02<br>Page 7 of 15 |
|---|---|

INSTRUCTIONS | CONTINGENCY ACTIONS

3.5 (continued)

**Facility 2**

- HV–203B, Fan F–21B exhaust damper is open.
- Fan F–21B, supply fan is running.
- HV–206B, Fan F–31B exhaust damper is open.
- Fan F–31B, exhaust fan is running.
- HV–212B, Fan F–32B exhaust damper is open.
- Fan F–32B, filter fan is running.
- HV–495, fresh air damper is closed.
- HV–496, exhaust air damper is closed.
- HV–497, cable vault exhaust damper is closed.

___3.6 VERIFY Containment Closure is established.

___3.7 Refer To MP-26-EPI-FAP06, "Classification and PARs," and DETERMINE reportability requirements.

Step 3.6, which instructs the operators to verify "Containment Closure", is after Step 3.5, which instructs the operators to place the CRACS in "Recirculation mode".

Level of Use **C**ontinuous

STOP THINK ACT REVIEW

Some Examinees felt that the order of the procedure steps *implies* that the actions directed before the Containment Closure step must be started within any existing time limit for the action to "VERIFY Containment Closure is established" ('original' correct answer to the question).

Q98
Additional Info
Provided

# MILLSTONE 2

**Lesson Title:** **Control Room Air Conditioning System**

**Revision:** **3/1**

**ID Number:** **CRA-00-C**



| | | |
|---|---|---|
| Submitted by: | John W. Riley | 4/2/2007 |
| | Developer | Date |
| Reviewed by: | Tim Grilley | 4/5/2007 |
| | Technical Reviewer | Date |
| Reviewed by: | N/A | N/A |
| | Cognizant Plant Supervisor *(Optional)* | Date |
| Approved by: | Trad Horner | 4/10/2007 |
| | Training Supervisor | Date |

| TEXT | ACTIVITIES/NOTES |
|---|---|
| The CRACS is equipped with the necessary dampers and controls for automatically switching to the full recirculation mode of operation for post-incident operation. System performance is continually monitored with alarms for high radiation, smoke, fan failure and excessive pressure drop through the filters. The Control Room Operator has remote, manual controls for positioning dampers, and operating fans and filters to ensure satisfactory control room conditions following an incident. | |
| To prevent infiltration from potentially contaminated areas within the plant, the Unit 2 Control Room should not be operated at a negative pressure with respect to adjacent areas in any mode of plant operation. | |
| **2. System Flow Paths** | |
| **a. Normal** | Figure 6 |
| Outside air (500 scfm) flows through the outside air inlet damper (2-HV-495) and the minimum outside air damper 2-HV-202 to the inlet plenum. This air is mixed with recirculated air from the exhaust fan through the open recirculation air damper 2-HV-209. The second path takes the air to the suction of the supply fans via the supply filters. After leaving the filters the supply fan provides additional motive force to move the air into the Control Room. The air leaves the Control Room and travels past the smoke detectors and on to the exhaust fan which keeps the air moving and exhaust to the Cable Vault. | PEO 2A |
| **b. Recirculation** | Figure 7 |
| In the Recirculation mode the air is drawn from the Control Room, past the smoke detectors by the exhaust fan. The air is discharged to the exhaust plenum. From here the air is directed to the suction of the filtration train fan via the PAC filters. The air then passes to the supply filters and on to the supply fans. From the supply fan the air is returned to the Control Room. Radiation Monitor 8011 sample fan is in operation to move a sample of the air past the radiation monitor. | PEO 2B |
| **c. Smoke Purge** | Figure 8 |
| During Smoke Purge air is drawn from the Control Room by the exhaust fan and discharged outside via 2-HV-208 and 2-HV-496. | PEO 2C |
| **d. Fresh Air Make-up** | Figure 9 |
| During Fresh Air Make-up air is drawn in from the outside via 2-HV-495 and 2-HV-211 and directed to the inlet of the PAC filters. The filter fan draws the air through the filters and discharges to the inlet of the supply filters. At the same time | PEO 2D |

| TEXT | ACTIVITIES/NOTES |
|---|---|
| **b. Normal Operation** | Figure 12 |
| During normal plant conditions, the Control Room Air Conditioning System is operated in the "Normal" mode. In the "Normal" mode one CRACS supply fan and one CRACS exhaust fan are required to be in operation. | PEO 4A |
| Outside air is introduced into the system at a rate of 500 scfm through the outside air supply louvers. This outside air flows through the outside air inlet damper (2-HV-495) and the minimum outside air damper (2-HV-202) to the suction plenum of the CRACS supply fans. This air is mixed with recirculated air from the CRACS exhaust fan. To balance the air flow through the system 500 scfm is exhausted to the Cable Vault through two open "discharge air to cable vault" dampers (2-HV-207 and 2-HV-497). Since the design flow of the CRACS supply and exhaust fans is 16,460 scfm, approximately 16,000 scfm of air must be recirculated back to the CRACS supply fan suction plenum through the open recirculation air damper (2-HV-209). | Figure 6 |
| 2-HV-209 is a manual damper that is maintained in the open position. This damper was changed to a manual damper. This was necessary due to single failure criteria and the potential loss of both CRACS trains, which could result from this damper changing position. | DM2-00-0356-05 |
| In this mode the Control Room Filtration Sub-system is in a standby status with the filter fans secured and the discharge dampers closed. | |
| For the CRACS to operate in the "Normal" mode the following switch positions are required on panels C-25A and C-25B:<br>• HS-8004C "Norm" position<br>• HS-8004D "Norm" position<br>• HS-8346 "OA" (outside air) position<br>• HS-8359 "OA" (outside air) position | |
| **2. Abnormal Operation** | |
| **a. Recirculation Mode** | Figure 7 |
| The CRACS Recirculation ("recirc") mode will be automatically initiated by any one of the following conditions:<br>1) Auxiliary Exhaust Actuation Signal (AEAS)<br>2) Enclosure Building Filtration Actuation Signal (EBFAS)<br>3) Intake Duct High Radiation Condition | PEO 4B<br>RO 3A |

| TEXT | ACTIVITIES/NOTES |
|---|---|
| 4) Radiation Monitor instrument failure or a loss of power to either instrument. | |
| An automatic initiation of the CRACS "recirc" mode will shift the CRACS to a complete recirculation mode of operation in which outside air is not introduced into the system and discharges to the cable vault and outside are isolated. Both filter fans start and their associated discharge dampers open. One supply fan and one exhaust fan will remain in operation with 16,640 scfm of air recirculating through the system and 2,500 scfm of the air flow passing through the Control Room Filtration Sub-system. | RO 3B |
| Filter Fans Inlet Damper, 2-HV-210 has had its control logic revised such that it will be open during normal and recirc modes. Damper 2-HV-210 will only be permitted to close by operator action during emergency fresh air intake mode of CRACS, when deemed necessary to purge the control room with outside air. | OP 2315A |
| Since the Control Room is at or near atmospheric pressure when the recirculation mode is initiated, in-leakage of unfiltered air should be insignificant. Assuming a pressure differential of 1/8 in. wg for openings and duct-work exposed to the environment external to the building and 1/16 in. wg for openings to external spaces, the calculated amount of unfiltered in-leakage should be less than 100 scfm. | |
| The CRACS can be manually shifted from "Normal" mode to "recirc" mode by placing both HS-8346 and HS-8359 to the "recirc" position for complete and redundant Control Room isolation. Single damper isolation for the Control Room may be attained by placing either one of the two control switches to "recirc" position. | |
| HS-8346 closes the following three dampers:<br>• minimum outside air (2-HV-202)<br>• discharge air to cable vault (2-HV-207)<br>• fresh air makeup damper (2-HV-211) | |
| HS-8359 closes the following two dampers:<br>• outside air inlet (2-HV-495)<br>• discharge air to cable vault (2-HV-497) | |
| When manually initiating the "recirc" mode, a filtration fan (F-32A or F-32B) must be started manually with its associated hand-switch on panels C-25A or C-25B. The "recirc air suction to the filter fans" damper (2-HV-210) will already be | |

| TEXT | ACTIVITIES/NOTES |
|---|---|
| open (due to the logic changes discussed above) when the filter fan is started. All other equipment will align itself in the same manner as the automatically initiated "recirc" mode. | |
| The automatic and manual initiation of the "recirc" mode involves completely redundant signals. Redundant radiation monitors, AEAS and EBFAS signals ensure that a single failure would not prohibit the Control Room from being automatically isolated. Also, two switches HS-8346 and HS-8359 provide redundant manual isolation of the Control Room. | |
| EBFAS or AEAS automatically shifts CRACS from normal mode to recirculation mode. If all control switches are left in their "normal" position, the CRACS automatically returns to normal mode when the EBFAS or AEAS is removed, with the exception of Fans F-32A and/or F-32B. These fans will remain running until operator action is taken to stop them. | |
| If ESAS has occurred and the signal can not be reset, all equipment must remain in actuated condition unless one of the following conditions exist:<br>• Personnel or equipment safety will be affected<br>• Specific operating procedures allow or require the signal to be overridden<br>• The equipment is not required for that mode | |
| **b. Smoke Purge Mode** | Figure 8<br>PEO 4C<br>RO 4A |
| Two smoke detectors (AE-8362 and AE-8358) are provided in the Control Room return duct-work to automatically isolate the supply unit and initiate purging operations. The Control Room is purged by operating the exhaust fans and discharging to atmosphere. Discharge dampers to the Cable Vault are automatically closed. Both Control Room Air Conditioning subsystems are served by the smoke detection system. | |
| If smoke is detected by the smoke detectors, both CRACS supply fans will receive a trip signal and the associated discharge damper for the running supply fan will automatically close. The exhaust air dampers (2-HV-208 and 2-HV-496) will receive an open signal and the discharge to cable vault (2-HV-207 and 2-HV-497) will receive a close signal. If the CRACS is in the "Recirculation" mode, the automatic smoke detection trip signal is overridden and the CRACS remains in "Recircualtion Mode" Thus, in the event of a fire during an accident condition, operator action would be required to shift or secure ventilation in order to prevent the spread of a fire. | RO 4B |

## TABLE 14.7.4 - 2 ASSUMPTION FOR FUEL HANDLING ACCIDENT IN CONTAINMENT

| | | |
|---|---|---|
| 17 | Control Room Emergency Filtered Reciculation Rate (from 1 hour after isolation) | 2,250 cfm |
| 18 | Control Room Free Volume | 35,656 $ft^3$ |
| 19 | Control Room Filter Efficiency (particulate/elemental/organic) | 90 / 90 / 70 % (1) |
| 20 | Dose Conversion Factors | Federal Guidance Reports 11 and 12 |

(1) 70% is a conservative analysis assumption for some iodine species. Technical Specifications can support assumptions for control room filter efficiencies of 90% for all iodine species.

**The following excerpt is from Technical Specification Bases 3.7.6:**

PLANT SYSTEMS

BASES

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room. For all postulated design basis accidents, the radiation exposure to personnel occupying the control room shall be 5 rem TEDE or less consistent with the requirements of 10 CFR 50.67

The Control Room Envelope (CRE) is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and other non-critical areas including adjacent support offices, and utility rooms. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, ceiling, ducting, valves, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

In order for the control room emergency ventilation systems to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

TS LCO 3.7.6.1 is modified by a footnote allowing the CRE boundary to be opened intermittently under administrative controls. This footnote only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

ACTIONS a., b., and c. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. ACTIONS d. and e. are applicable in MODES 5 and 6, or when recently irradiated fuel assemblies are being moved. The control room emergency ventilation system is required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis), due to radioactive decay. The term "recently" is defined as all irradiated fuel assemblies, until analysis is performed to determine

a specific time. The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm ± 10%.

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), will require operator action to return to a full train lineup. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train lineup. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until 1 hour into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day allowed outage time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day allowed outage time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary. Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

Surveillance Requirement 4.7.6.1.c.1 dictates the test frequency, methods and acceptance criteria for the Control Room Emergency Ventilation System trains (cleanup trains). These criteria all originate in the Regulatory Position sections of Regulatory Guide 1.52, Rev. 2, March 1978 as discussed below.

Section C.5.a requires a visual inspection of the cleanup system be made before the following tests, in accordance with the provisions of section 5 of ANSI N510-1975:

- in-place air flow distribution test
- DOP test
- activated carbon adsorber section leak test

**The following excerpt is from LBDCR #04-MP2-011, AR #07003671-04:**



dose will be retained with the acceptance criteria remaining as "substantially below 10 CFR Part 100 guidelines".

The proposed licensing and plant operational changes are discussed in Section 2.0. These changes require appropriate changes to the MPS2 operating license and technical specifications, which are also described in Section 2.0 of this report. The changes considered as a result of the re-analyses and clarifications are listed below:

a. Revise definition of Dose Equivalent I-131 in Section 1.19 of the Technical Specifications Definitions to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988, as the source of inhalation thyroid dose conversion factors (Reference 8).

b. Change the combined leakage rate acceptance criteria for all penetrations that are secondary containment bypass leakage paths in Technical Specification LCO 3.6.1.2, "Containment Leakage" from < 0.0072 $L_a$ to < 0.014 $L_a$.

c. Change the control room unfiltered in-leakage limit in Technical Specification SR 4.7.6.1.e.3 from 130 SCFM to 200 SCFM and clarify the applicability.

d. Revise the primary to secondary leakage limit in Technical Specification LCO 3.4.6.2.c from 0.035 GPM to 75 GPD through any one steam generator.

e. Approve a new operator action to isolate the control room up to 4 hours after a MSLB.

f. Approve a timing change to an existing operator action such that within 1 hour (instead of 10 minutes) of isolation, the control room emergency ventilation system is placed in filtered recirculation.

g. Delete Technical Specification LCO 3.3.4 and SR 3.3.4, "Containment Purge Valve Isolation Signal," since its intent is addressed by LCO 3.6.3.2, "Containment Ventilation System."

h. Revise the language of Technical Specification LCO 3.3.3.1 Action b to be consistent with similar action statements within that technical specification.

i. Revise the technical specification bases to reflect the above listed changes, as appropriate, in accordance with Technical Specification 6.23, "Bases Control Program."

# Regulatory Correspondence Distribution List

Document Title: Issuance of Amendment Regarding Alternate Source Term - Millstone Unit 2
Serial Number: 07-0462
**Requested Action:**
**Action Assigned To:** Sta. Lic.
**Date Due to NLOS:** Licensing POC: B. Bartron
**Date Due To NRC:**

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 31, 2007

SERIAL # 07-0462

REC'D JUN 11 2007

NUCLEAR LICENSING

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO.2 - ISSUANCE OF AMENDMENT REGARDING ALTERNATE SOURCE TERM (TAC NO. MD2346)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 298 to Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated June 13, 2006, as supplemented by letter dated March 6, 2007.

This amendment revises the MPS2 licensing basis in the area of radiological dose analysis for design-basis accidents using the alternative source term permitted by Title of the *Code of Federal Regulations* 50.67, "Accident source term". Additionally, the amendment revises the MPS2 Technical Specifications to be consistent with the amended licensing-basis.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

John D. Hughey

John Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 298 to License No. DPR-65
2. Safety Evaluation

cc w/encls: See next page

Millstone Power Station, Unit No. 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 298
License No. DPR-65

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Dominion Nuclear Connecticut, Inc. (the licensee) dated June 13, 2006, as supplemented by letter dated March 6, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 298 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. Dominion Nuclear Connecticut, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to License No. DPR-65
and the Technical Specifications

Date of Issuance: May 31, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 298

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace page 3 of License No. DPR-65 with the attached revised page 3.

Replace the following pages of the Appendix A, Technical Specifications with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| V | V |
| XI | XI |
| 1-4 | 1-4 |
| 3/4 3-24 | 3/4 3-24 |
| 3/4 3-36 | 3/4 3-36 |
| 3/4 3-37 | 3/4 3-37 |
| 3/4 4-9 | 3/4 4-9 |
| 3/4 6-2 | 3/4 6-2 |
| 3/4 7-16a | 3/4 7-16a |
| 3/4 7-18 | 3/4 7-18 |

Connecticut, in accordance with the procedures and limitations set forth in this renewed operating license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2700 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix, as revised through Amendment No. 298 , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS


| SECTION | | PAGE |
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| 3/4.2 POWER DISTRIBUTION LIMITS | | |
| 3/4.2.1 | LINEAR HEAT RATE | 3/4 2-1 |
| 3/4.2.2 | Deleted | |
| 3/4.2.3 | TOTAL INTEGRATED RADIAL PEAKING FACTOR - $F_r^T$ | 3/4 2-9 |
| 3/4.2.4 | AZIMUTHAL POWER TILT | 3/4 2-10 |
| 3/4.2.5 | Deleted | |
| 3/4.2.6 | DNB MARGIN | 3/4 2-13 |
| 3/4.3 INSTRUMENTATION | | |
| 3/4.3.1 | REACTOR PROTECTIVE INSTRUMENTATION | 3/4 3-1 |
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| | Remote Shutdown Instrumentation | 3/4 3-28 |
| | Accident Monitoring | 3/4 3-31 |
| 3/4.3.4 | Deleted | |
| 3/4.4 REACTOR COOLANT SYSTEM | | |
| 3/4.4.1 | COOLANT LOOPS AND COOLANT CIRCULATION | 3/4 4-1 |
| | Startup and Power Operation | 3/4 4-1 |
| | HOT STANDBY | 3/4 4-1a |
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| | COLD SHUTDOWN - Reactor Coolant System Loops Filled | 3/4 4-1d |
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## INDEX

BASES


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| 3/4.1 REACTIVITY CONTROL SYSTEMS | | |
| 3/4.1.1 | BORATION CONTROL | B 3/4 1-1 |
| 3/4.1.2 | Deleted | |
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| 3/4.2 POWER DISTRIBUTION LIMITS | | |
| 3/4.2.1 | LINEAR HEAT RATE | B 3/4 2-1 |
| 3/4.2.2 | Deleted | |
| 3/4.2.3 | TOTAL INTEGRATED RADIAL PEAKING FACTOR - $F_r^T$ | B 3/4 2-1 |
| 3/4.2.4 | AZIMUTHAL POWER TILT | B 3/4 2-1 |
| 3/4.2.5 | Deleted | |
| 3/4.2.6 | DNB MARGIN | B 3/4 2-2 |
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| 3/4.3.2 | ENGINEERED SAFETY FEATURE INSTRUMENTATION | B 3/4 3-1 |
| 3/4.3.3 | MONITORING INSTRUMENTATION | B 3/4 3-2a |
| 3/4.3.4 | Deleted | |

AZIMUTHAL POWER TILT - $T_q$

1.18 AZIMUTHAL POWER TILT shall be the difference between the maximum power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{AZIMUTHALPOWERTILT} = \left[\frac{\text{Maximum power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro-curie/gram) which alone would produce the same CEDE-thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed under inhalation in Federal Guidance Report No. 11 (FGR11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$ shall be the average sum of the beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subinterval, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With the number of OPERABLE channels less than the number of MINIMUM CHANNELS OPERABLE in Table 3.3-6, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 DELETED

4.3.3.1.3 Verify the response time of the control room isolation channel at least once per 18 months.

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REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 75 GPD primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE shall be demonstrated to be within limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $< L_a$, 0.50 percent by weight of the containment air per 24 hours at $P_a$, 54 psig.

b. A combined leakage rate of $< 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests when pressurized to $P_a$.

c. A combined leakage rate of $< 0.014\ L_a$ for all penetrations that are secondary containment bypass leakage paths when pressurized to $P_a$.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, or (c) with the combined bypass leakage rate exceeding $0.014\ L_a$, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with the Containment Leakage Rate Testing Program.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (continued)

MODES 5 and 6, or during irradiated fuel movement within containment or the spent fuel pool:**

d. With one Control Room Emergency Ventilation Train inoperable, restore the inoperable train to OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation Train in the recirculation mode of operation, or immediately suspend CORE ALTERATIONS, and the movement of irradiated fuel assemblies.

e. With both Control Room Emergency Ventilation Trains inoperable, or with the OPERABLE Control Room Emergency Ventilation Train required to be in the recirculation mode by ACTION d. not capable of being powered by an OPERABLE normal and emergency power source, immediately suspend CORE ALTERATIONS, and the movement of irradiated fuel assemblies.

** In MODES 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then ACTION 3.7.6.1.d or 3.7.6.1.e shall be invoked as applicable.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that control room air in-leakage is less than 200 SCFM with the Control Room Emergency Ventilation System operating in the recirculation/filtration mode.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

Final Submittal for RO Question #35: Rev 0 Received in Regen I 3/23/09

The plant was operating at 100% power with all equipment in a normal alignment. Bus 24C is supplying Bus 24E. The "C" RBCCW Pump suddenly trips on fault. The Balance of Plant (BOP) operator starts the "B" RBCCW Pump to supply Facility 2 and places the 'C' RBCCW Pump handswitch in Pull-To-Lock. NO other operator action is taken.

Then, a circuit failure in the Facility 2 ESAS Actuation Cabinet causes an inadvertent actuation of SIAS, CIAS, EBFAS and UV
ALL plant systems and components function as designed (including those driven by the Facility 2 inadvertent actuation).

Which of the following correctly describes the condition of the RBCCW Pumps and Heat Exchanger TCVs based on the inadvertent ESAS actuation?

☐ **A** The 'A' and 'B' RBCCW Pumps are running.
'A' heat exchanger TCV throttling on temperature.
'B' and 'C' heat exchanger TCVs full open.

☐ **B** The 'A' and 'B' RBCCW Pumps are running.
'A' and 'B' heat exchanger TCVs throttling on temperature.
'C' heat exchanger TCVs full open.

☑ **C** Only the 'A' RBCCW Pump is running.
'A' heat exchanger TCV throttling on temperature.
'B' and 'C' heat exchanger TCVs full open.

☐ **D** Only the 'A' RBCCW Pump is running.
'A' and 'B' heat exchanger TCVs throttling on temperature.
'C' heat exchanger TCVs full open.

---

Submitted Answer Justification:

C is correct. The 'B' RBCCW Pump SIAS/LNP Block handswitch at the breaker was left in BLOCK which will NOT allow the 'B" RBCCW Pump to start on SIAS. A Fac. 2 SIAS will cause the 'B' (swing) and 'C' RBCCW heat exchanger TCVs to get a full open signal.

A is incorrect. The 'B' RBCCW Pump will NOT start on SIAS due to the SIAS/LNP Block handswitch being left in the BLOCK position.

B is incorrect. The 'B' RBCCW Pump will NOT start on SIAS due to the SIAS/LNP Block handswitch being left in the BLOCK position. Additionally, the 'B' RBCCW heat exchanger is the swing heat exchanger and its TCV will get an open signal on a SIAS from either facility of ESAS.

D is incorrect. While it is true that only the 'A' RBCCW Pump will start on the SIAS, the 'B' RBCCW heat exchanger is the swing heat exchanger and its TCV will get an open signal on a SIAS from either facility of ESAS.

**Licensee's Justification For Change For RO Question #35:**

The original correct answer for this question was choice "A". The original question was changed from "*a small break LOCA followed by a plant trip and Loss of Offsite Power*" to "*an inadvertent SIAS, CIAS, EBFAS, and UV on Facility*". In the original version, the "B" RBCCW Pump is deenergized (momentary loss of Bus 24C, which is tied to Bus 24E), and will NOT restart because its SIAS/LNP Block switch is still in the "BLOCK" position.

Original draft version of RO Question #35:

> *The plant was operating at 100% power with all equipment in a normal alignment. Bus 24C is supplying Bus 24E. The "C" RBCCW Pump suddenly trips on fault. The Balance of Plant (BOP) operator starts the "B" RBCCW Pump to supply Facility 2 and places the 'C' RBCCW Pump handswitch in Pull-To-Lock. NO other operator action is taken.*
>
> *Subsequently, a small break LOCA occurs followed by a plant trip and Loss Of Offsite Power (LNP) SIAS, CIAS, EBFAS, MSI and UV actuate and all equipment functions as designed.*
>
> *Which of the following correctly describes the condition of the RBCCW Pumps and Heat Exchanger TCVs following this event?*

In the final submitted version of the question (see previous page), even though the "B" RBCCW Pump is supplying Facility 2 components, it remains energized after the Facility 2 UV Undervoltage (UV) because it is still powered by Facility 1 (Bus 24E aligned to Bus 24C), which is NOT deenergized on an inadvertent Facility 2 UV. Because the "B" RBCCW Pump remains energized, the position of the SIAS/LNP Block switch is irrelevant (With the SIAS/LNP Block switch in the BLOCK position, the "B" RBCCW Pump will not restart on SIAS or LNP).

Choice "A" states, "*The 'A' and 'B' RBCCW Pumps are running.*"
Choice "C" states, "*Only the 'A' RBCCW Pump is running.*"
The remaining portions of choice "A" and choice "C" are identical.
Choices "B" and "D" are incorrect because the "B" heat exchanger TCV will NOT throttle on temperature after an inadvertent SIAS on Facility 1 or Facility 2. (i.e., a SIAS on either facility will cause the "B" RBCCW Heat Exchanger TCV to go full open.)
Consequently, choice "A" is correct and all other choices are incorrect.

Additional documentation is attached.

Final Submittal for SRO Question #77 (#2):

While performing a plant cooldown, the crew was about to place Shutdown Cooling (SDC) in service with "A" and "B" Reactor Coolant Pumps (RCPs) in service, when Pressurizer Pressure Narrow Range Instrument, P-103-1, suddenly fails high.

Which of the following describes the impact of this failure and the action that the US must direct to stabilize the plant?

☑ **A** The "A" Power Operated Relief Valve (PORV) will open and RCS pressure will rapidly lower. Place the "A" LTOP switch to "High", close PORV Block Valve, secure the "A" and "B" RCPs, and place SDC in service.

☐ **B** The Facility 1 Safety Injection Tank (SIT) outlet Valves will open causing the SITs to inject. Override and close the SIT Outlet Valves and restore Pressurizer level to the desired setpoint.

☐ **C** SDC Suction Isolation Valve, SI-652, will NOT open until the high pressure input is cleared. Direct I&C to bypass or disconnect Pressurizer Pressure Narrow Range Instrument, P-103-1.

☐ **D** The "A" Power Operated Relief Valve (PORV) will open and RCS pressure will rapidly lower. Immediately place the associated PORV Block Valve to 'Close' and log into the LTOP Technical Specification LCO.

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Submitted Answer Justification:

A is correct. In this alignment, the LTOP switch is in the LOW position, resulting in the low pressure setpoint being applied to the PORVs. If Pressurizer Pressure Narrow Range Instrument, P-103-1, suddenly fails high, the "A" PORV will open resulting in RCS pressure rapidly lowering. The PORV can only be closed by placing the "A" LTOP switch in the HIGH position. The rapid reduction in RCS pressure will result in the loss of NPSH for the RCPs, so they must be secured. With the loss of the RCPs, core cooling must be established by initiating SDC.

B is incorrect. Although the Facility 1 SIT Outlet Valves get an open signal from the failed pressure instrument, the present plant alignment requires the pressure input to be bypassed to prevent inadvertent opening.

C is incorrect. The high pressure input to the SDC Suction Isolation Valves has been removed. Additionally, P-103-1 (Facility 1) does NOT align with SI-652 (Facility 2).

D is incorrect. The "B" PORV will NOT open as it is NOT associated with P-103-1. Additionally, if P-103 failed high, attempting to close the "B" PORV would NOT work because it is interlocked open with the "B" LTOP switch set to LOW.

**Licensee's Justification For Change For SRO Question #77 (#2):**

The original correct answer for this question is choice "A". The original draft answer did not include the phrase, "*and place SDC in service.*" In-house validation suggested that this phrase be added to complete 'the action that the US must direct to stabilize the plant'; however, SDC cannot be placed in service with Pressurizer Pressure instrument, P-103-1, failed high. As a result of the instrument failure, SDC Isolation Valve, SI-651, will be interlocked closed, preventing SDC from being placed in service. This error was not recognized during subsequent validation.

The stem of the question asks what must be done "*to stabilize the plant.*" While the actions to "*place SDC in service*" would certainly stabilize the plant, the time required to do so to meet choice "A" are unrealistic. Manual operation of Shutdown Cooling Isolation Valve, SI-651 in Containment would far exceed allowable radiological dose and the time necessary to determine inputs to the valve from P-103-1 would require in-depth circuit analysis since there are no procedures readily available to override or defeat this interlock.

The question does not indicate what RCS temperature is at the time of the instrument failure. Therefore, in this situation, once RCPs are secured it can be assumed that the cooldown would be stopped and RCS temperature stabilized using the steam generators while natural circulation develops. This was not a selection option for this question.

If Shutdown Cooling Isolation Valve, SI-651, cannot be opened due to the failure of Pressurizer Pressure instrument, P-103-1, then, obviously, Shutdown Cooling cannot be placed in service to meet choice "A"; therefore, choice "A" is incorrect. All other answers have been validated as being incorrect; therefore; SRO question #77 (2) has NO correct answer.

Additional documentation is attached.

**The following highlighted excerpts from the Shutdown Cooling lesson plan, SDC-00-C, demonstrate that Shutdown Cooling Isolation Valve, SI-651, cannot be opened with Pressurizer Pressure Instrument, P-103-1, failed above 280 psia.**

| **d. Control and Instruments** | |
|---|---|
| The Suction Line Isolation Valve motor operators are powered from MCC B61 (SI-652) and MCC B51 (SI-651).<br><br>The 480V power supply to 2-SI-652 is normally isolated in order to prevent inadvertent opening and subsequent system over pressurization during power operation. This isolation switch is located in the Control Room behind panel C80. Operation of this switch will actuate annunciator D-39 on C-01 whenever power is made available to 2-SI-652.<br><br>To address single-failure concerns for post-accident boron precipitation, 2-SI-651 is equipped with an alternate power supply from 480 volt MCC B61. An administratively controlled kirk-key shifts control to local panel and bypasses the RCS pressure interlock (280 psia). | PEO-4B |
| The control switches for SI-651 and SI-652 are located in the Control Room on panel C-01. They are two position OPEN/CLOSE keylock switches. Each of the valves has RED (open) and GREEN (closed) indication lights. | RO-2A |
| Procedurally, the LPSI system, SDC heat exchangers, and interconnecting piping can not be exposed to an RCS pressure greater than 265 psia or temperature greater than 300° F. To protect the piping against excessive pressure SI-651 and SI-652 are interlocked with the low range Pressurizer pressure channels (P-103 and P-103-1). The range of these instruments is 0 to 1600 psia. Pressure is displayed on C-03 and on C-21 and can also be monitored on the Plant Process Computer. The SDC Suction Isolation Valves are interlocked closed if RCS pressure is greater than 280 psia. If SI-651 or SI-652 were already open (i.e. SDC in service) and pressure rose above 280 psia an alarm is received on C01 alerting the operators. | |
| Each Suction Isolation Valve motor operator has a manual handwheel and clutch mechanism allowing local operation of the valve. | |

## Attachment 3 - Interlocks & Automatic Features

| Instrument # | Feature |
|---|---|
| LT-3001<br>LT-3002<br>LT-3003<br>LT-3004 | Two out of four Refueling Water Storage Tank (RWST) level indications less than setpoint will initiate a Sump Recirculation Actuation Signal (SRAS) from the Engineered Safeguards Actuation System (ESAS).<br>Provided ESAS were not bypassed and a signal were to occur, the following Shutdown Cooling System components would be affected:<br>• LPSI Pump "A" and "B" would trip;<br>• SI/CS Minimum flow recirc valves (2-SI-659 & 2-SI-660) would close (provided that HS-3659A and HS-3660A are in the "OPER" position);<br>• Containment Sump Outlet Valves (CS-16.1 A/B) open.<br>• Shutdown Cooling (SDC) Heat Exchanger RBCCW outlet valves would open (RB-13.1 A/B). |
| PT-102A<br>PT-102B<br>PT-102C<br>PT-102D | Two out of four Pressurizer Pressure indications less than setpoint will initiate SIAS from the Engineered Safeguards Actuation System (ESAS). SIAS is blocked during SDC operation, but if it was not blocked, the following components within the LPSI system or affecting LPSI system performance would be effected:<br>• LPSI Pump "A" and "B" start.<br>• LPSI injection valves (2-SI-615, 2-SI-625, 2-SI-635, & 2-SI-645) open.<br>• ESF Room Fans (F-15A & F-15B) start. |
| PT-103 | Prevents the shutdown cooling suction line isolation valve 2-SI-652 from being opened if a high Reactor Coolant System (RCS) pressure condition exists (setpoint: 280 psia increasing). |
| PT-103-1 | Prevents the shutdown cooling suction line isolation valve 2-SI-651 from being opened if a high Reactor Coolant System (RCS) pressure condition exists (setpoint: 280 psia increasing). |

Final Submittal for SRO Question #98 (#23):

A Fuel Handling Accident has occurred inside Containment as evidenced by several Containment process and area radiation monitors rising or in alarm. All personnel were immediately evacuated.

In order to limit personnel exposure, the Shift Manager must ensure which of the following are performed?

☑ **A** The Containment (Outage) Equipment Hatch must be closed no more than 30 minutes after the event.

☐ **B** Containment Purge valves, AC-4, 5, 6, and 7 must be closed no more than 50 minutes after the event.

☐ **C** The Transfer Tube Isolation Valve, RW-280, must be closed no more than 50 minutes after the Transfer Carriage is in the SFP

☐ **D** One train of CRACS must be operating in Recirculation mode no more than 30 minutes after the event.

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Submitted Answer Justification:

A is correct. Per AOP-2577 (Fuel Handling Accident), Rev. 008-02, Section 1.0, Containment must be isolated within 30 min. of the event.

B is incorrect. The Containment Purge Valves would be closed for a Fuel Handling Accident in Containment, but they would likely have closed automatically. However, if the auto actuation has not occurred, they must be manually closed immediately.

C is incorrect. The Transfer Tube Isolation Valve, RW-280, is not included in the components that must be closed for Containment Isolation in a Fuel Handling Accident as the opening is low in the refuel pool. This valve would be closed for a loss of Refuel Pool level and takes about 30 minutes to close.

D is incorrect. The calculation for the Control Room radiological exposure following a Fuel Handling Accident is based on having at least one Control Room Air Conditioning train operating in the Recirculation mode within 60 minutes of the event. The calculations used to require recirc mode within 10 minutes. However, new calculations recently changed the requirement to 50 minutes with a 10 minute penalty for other system alignment complications. (AOP-2577 had NOT yet been updated when the question was written.)

**Licensee's Justification For Change For SRO Question #98 (#23):**

The original correct answer for this question is choice "A"; however, given that the stem of the question asks the examinee what must be done "*to limit personnel exposure*", it could be argued that choice "D" is also correct. Choice "D" states, "*One train of CRAC must be operating in Recirculation mode no more than 30 minutes after the event.*" This is not an incorrect action in order to limit personnel exposure.

In 2007, Millstone Unit 2 (MPS2) implemented License Amendment No. 298 to Technical Specifications (TAC NO. 2346). This amendment revised the MPS2 licensing basis in the area of radiological dose analysis for design-basis accidents using the alternative source term (AST) as permitted by 10CFR50.67.

Full implementation of the AST at MPS2, as defined in Section 1.2.1 of RG 1.183, required re-analysis of the following accidents: LOCA, Fuel Handling Accident (FHA), Cask Tip Accident, SGTR, MSLB, CREA. In all of the re-analysis performed, for accidents which required the Control Room Air Conditioning System (CRACS) to be placed in the filtered, recirculation mode, the required time for operator action was changed from the previous 10 minute requirement to "within 1 hour".

AOP 2577, "Fuel Handling Accident," includes as part of the 'Discussion' section the following statement:

> "*Control Room radiological exposure following a fuel handling accident in containment is calculated on having at least one train of CRACS operating in the recirculation mode within 10 minutes of the accident.*"

This statement within AOP 2577 is only a statement of fact from the previous requirement before implementation of the AST approach at MPS2. Procedure step 3.5 in AOP 2577 includes actions necessary to place the CRAC System in filtered recirculation; however, there is no specified time requirement for implementation of this step.

Prior to attending license school, the SRO Upgrade candidates attended a presentation (C07601), as part of Licensed Operator Requalification Training concerning the Alternate Source Term and the affect on Control Room Air Conditioning (CRAC) System and the time change to "within 1 hour" for filtered recirculation.

Additionally, OP 2260, "EOP User's Guide" was revised to reflect the re-analysis and implementation of the AST. The time chosen was a conservative 50 minutes (rather than 1 hour by the accident analysis), to place the CRAC System in filtered recirculation for LOCA, SGTR, and MSLB accidents.

Since placing the CRAC System in filtered recirculation mode is credited as part of the accident analysis for FHA to limit personnel exposure, and given the absence of any procedure directed time requirement within AOP 2577, it can be argued that choice "D" is also an acceptable answer, since 30 minutes is "within 1 hour" as outlined in the accident analysis. Therefore, examinees should be given credit for both choice "A" and choice "D".

A post-exam Condition Report (CR325747) has been generated to identify the modification required for AOP 2577, "Fuel Handling Accident."

Additional documentation is attached.