September 26, 2008

Mr. James A. Spina, Vice President
Calvert Cliffs Nuclear Power Plant, Inc.
Calvert Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT - NRC EXAMINATION REPORT 05000317/2008301 AND 05000318/2008301

Dear Mr. Spina:

On June 25, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an examination at the Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2. The enclosed report documents the examination findings, which were discussed on September 3, 2008, with Mr. W. Holston and other members of your staff.

The examination included the evaluation of four applicants for reactor operator licenses, four applicants for instant senior operator licenses and three applicants for upgrade senior operator licenses. The written and operating examinations were developed using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The license examiners determined that five of the eleven applicants satisfied all of the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

No findings of significance were identified during this examination. However, it was determined that the quality of the submitted examination material was not within the range of acceptability expected by the NRC. Greater than 20% of the written and operating test items were considered to be unsatisfactory.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at [http://www.nrc.gov/reading-rm/adams.html](http://www.nrc.gov/reading-rm/adams.html) (the Public Electronic Reading Room).

Sincerely,

***/RA/***
Samuel L. Hansell, Jr., Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-317, 50-318
License Nos. DPR-53, DPR-69

Enclosure: NRC Examination Report 05000317/20008301 and 05000318/2008301

Mr. James A. Spina
Vice President
Calvert Cliffs Nuclear Power Plant
Constellation Generation Group, LLC
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

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cc w/encl:
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J. Gaines, Director, Licensing, Calvert Cliffs Nuclear Power Plant
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# EXAMINATION REPORT

## U.S. NUCLEAR REGULATORY COMMISSION

## REGION I

| | |
|---|---|
| Dockets: | 50-317, 50-318 |
| Licenses: | DPR-53, DPR-69 |
| Report: | 05000317/2008301; 05000318/2008301 |
| Licensee: | Constellation Generation Group |
| Facility: | Calvert Cliffs Nuclear Power Plant, Units 1 and 2 |
| Location: | Lusby, MD |
| Dates: | June 16 – August 22, 2008 |
| Inspectors: | D. M. Silk, Chief Examiner, Operations Branch<br>J. M. D'Antonio, Senior Operations Engineer<br>B. C. Haagensen, Operations Engineer |
| Approved By: | Samuel L. Hansell, Jr., Chief<br>Operations Branch<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

ER 05000317/2008301, 05000318/2008301; June 16 – August 22, 2008; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of four applicants for reactor operator licenses, four applicants for instant senior operator licenses and three applicants for upgrade senior operator licenses at the Calvert Cliffs Nuclear Power Plant Units 1 and 2. The facility licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The written examination was administered by the facility on June 25, 2008. NRC examiners administered the operating tests on June 16 – 20, 2008. The license examiners determined that five of the eleven applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

The examiners reviewed all eleven license applications submitted by the licensee to ensure the applications reflected that each applicant satisfied relevant license eligibility requirements. The applications were submitted on NRC Form 398, "Personal Qualification Statement," and NRC Form 396, "Certification of Medical Examination by Facility Licensee." The examiners also audited one of the license applications in detail to confirm that it accurately reflected the subject applicant's qualifications. This audit focused on the applicant's experience and on-the-job training, including control manipulations that provided significant reactivity changes.

b. Findings

No findings of significance were identified.

.2 Operator Knowledge and Performance

a. Examination Scope

On June 25, 2008, the licensee proctored the administration of the written examinations to all eleven applicants. The licensee staff graded the written examinations based upon the original answer key, analyzed the results, and presented their analysis to the NRC on July 29, 2008. Due to numerous post-examination comments, the NRC graded the written examinations based upon a revised answer key.

The NRC examination team administered the various portions of the operating examination to all eleven applicants on June 16 – 20, 2008. The four reactor operator applicants participated in two dynamic simulator scenarios, a control room and facilities walkthrough test consisting of eleven system tasks, and an administrative test consisting of four administrative tasks. The four instant senior operator applicants participated in two dynamic simulator scenarios, a control room and facilities walkthrough test consisting of ten system tasks, and an administrative test consisting of five administrative tasks. The two upgrade senior operator applicants participated in two dynamic simulator scenarios, a control room and facilities walkthrough test consisting of five system tasks, and an administrative test consisting of five administrative tasks. A third upgrade applicant was administered the same test items as the other two upgrade applicants except for the in-plant portion of the facilities walkthrough. At the time of the operating examination this upgrade applicant did not have access to the protected area.

b. Findings

Ten of the eleven applicants achieved passing scores on all parts of the operating test. Due to the protected area access issue, the one upgrade applicant was not able to complete all portions of the operating test. This individual was not able to achieve an overall passing score of the operating examination based upon the test items that he was able to take. Therefore, this applicant still needs to complete the in-plant portion of the operating test. The SRO upgrades license application status will be held in suspension until the in-plant portion of the exam is completed, or it is determined that this portion of the exam cannot be administered in a reasonable time frame.

Two reactor operator and three senior reactor operator applicants failed their written examinations. For the written examinations, the reactor operator applicants' average scores were 76.5 percent and ranged from 64.7 to 80.9 percent. The senior operator applicants' overall average scores were 81.9 percent and ranged from 78.0 to 86.8 percent. The overall written examination average for both license levels was 79.9 percent. Chapter ES-403 and Form ES-403-1 of NUREG 1021 require the licensee to analyze the validity of any written examination questions that were missed by half or more of the applicants. The licensee conducted this performance analysis for 25 questions that met this criterion and submitted the analysis to the chief examiner. The licensee modified lesson plans as appropriate to address applicant performance issues and conducted remediation training.

The licensee submitted post-examination comments for 13 questions. The licensee recommended deleting four questions and accepting two answers for nine questions. After reviewing the licensee's comments, the NRC decided to delete nine questions, accept two responses for two questions, and make no change to the remaining two questions. See Attachment 5 for a summary of the licensee comments and NRC responses. Attachment 6 contains the details of the licensee's comments for each question and the NRC's responses. As per ES-501 C.2.c, the examiners reviewed the sample plan in light of the deleted questions. Even with the numerous deletions, every knowledge and ability (K/A) category has been sampled by the remaining questions.

The written examination resulted in a high failure rate (~45%). This issue has been captured in the site corrective action program under Condition Report IRE-032-618. A root cause evaluation was conducted. The root cause analysis team consisted of licensee staff, Constellation fleet personnel, and individuals from other licensees. The root cause was determined to be inadequate supervisory and management oversight in the examination preparation phase. Contributing causes were determined to be:
1) insufficiently systematic validation process, 2) inadequate fundamental knowledge of regulatory standards, and 3) inadequate treatment of existing operating experience. In the fourth quarter of 2008, the NRC has scheduled a problem identification and resolution inspection to review the licensee's findings and the corrective actions regarding their root cause analysis report.

.3 Initial Licensing Examination Development

a. Examination Scope

The licensee staff developed the examinations in accordance with NUREG-1021, Revision 9, Supplement 1. All licensee facility training and operations staff involved in examination preparation and validation were listed on a security agreement. The licensee submitted both the written and operating examination outlines on March 24, 2008. The chief examiner reviewed the outlines against the requirements of NUREG-1021, Revision 9, Supplement 1, and provided comments to the licensee. The licensee submitted the draft examination package on April 28, 2008. The chief examiner reviewed the draft examination package against the requirements of NUREG-1021, Revision 9, Supplement 1, and provided comments to the licensee on April 30, 2008, and May 2, 2008, regarding the operating and written examinations, respectively. The NRC conducted an onsite validation of the operating examinations and provided further comments during the week of May 19, 2008. The licensee satisfactorily completed comment resolutions on June 3 and June 12, 2008, for the operating and written examinations, respectively.

b. Findings

The NRC approved the initial examination outline and advised the licensee to proceed with the operating examination development.

The examiners determined that the operating examination initially submitted by the licensee was unsatisfactory as greater than 20% of the job performance measures were replaced and the simulator scenarios required enhancements. Initially, the scenario set was developed so that most applicants would be evaluated in three scenarios. Later, it was decided that only two scenarios were necessary to evaluate the applicants. Thus, the scenarios required additional malfunctions to ensure that each applicant could be properly evaluated. The NRC also determined that the written examination was unsatisfactory as greater than 20% of the questions for both the reactor operator and senior reactor operator examinations were unsatisfactory. See Attachment 4 for details regarding the unsatisfactory test items.

.4 Simulation Facility Performance

a. Examination Scope

The examiners observed simulator performance with regard to plant fidelity during the examination validation and administration.

b. Findings

No findings of significance were identified.

.5 Examination Security

a. Examination Scope

The examiners reviewed examination security for examination development, validation, and administration for compliance with NUREG-1021 requirements. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The chief examiner presented the examination results to Mr. W. Holston, Training Manager, and other members of the licensee's management staff on September 3, 2008. The licensee acknowledged the findings presented.

The licensee did not identify any information or materials used during the examination as proprietary.

ATTACHMENTS
ATTACHMENT 1: SUPPLEMENTAL INFORMATION
ATTACHMENT 2: SIMULATOR FIDELITY REPORT
ATTACHMENT 3: DOCUMENTS REVIEWED DURING THE EXAMINATION PROCESS
ATTACHMENT 4: UNSATISFACTORY TEST ITEMS
ATTACHMENT 5: SUMMARY OF LICENSEE POST-EXAM COMMENTS & NRC RESPONSES
ATTACHMENT 6: LICENSEE COMMENTS AND NRC RESPONSES

## ATTACHMENT 1

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

J. Holmes, Nuclear Training Instructor
W. Holston, Training Director
N. Lavato, Operations Continuing Training Supervisor
R. Pace, Operations Training Manager
M. Wasem, Initial Licensed Operator Training Supervisor

NRC Personnel

S. Kennedy, Senior Resident Inspector

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Closed

NONE

Discussed

NONE

## ADAMS DOCUMENTS REFERENCED

Accession No. ML082480504 – FINAL-Written Exams with Answer Keys
Accession No. ML082210541 – FINAL-Operating Exam Section A
Accession No. ML082210551 – FINAL-Operating Exam Section B
Accession No. ML082210558 – FINAL-Operating Exam Section C
Accession No. ML082320241 – Post Exam Comments
Accession No. ML082670292 – Post Exam Comments - Addendum

## ATTACHMENT 2
## SIMULATOR FIDELITY REPORT

| ES-501 | Simulator Fidelity Report | Attachment 2 |
|---|---|---|

Facility Licensee: Calvert Cliffs
Facility Docket No.: 50-317 & 50-318
Operating Test Administered on: 6/16-20/08

*This form is to be used only to report observations. These observations do not constitute audit or inspection findings and, without further verification and review in accordance with IP 71111.11, are not indicative of noncompliance with 10 CFR 55.46. No licensee action is required in response to these observations.*

While conducting the simulator portion of the operating tests, examiners observed the following items:

| Item | Description |
|---|---|
| The simulator would not properly reset following a dynamic simulator session. | The simulator plant process computer failed upon a scenario reset. The licensee's trouble-shooting revealed a degraded 15 volt power supply. This power supply was replaced after a three and a half hour delay. The dynamic simulator sessions were resumed without any other simulator problems. |
| | |

**ATTACHMENT 3**
**DOCUMENTS REVIEWED DURING THE EXAMINATION PROCESS**

Condition Reports:
- IRE-031-349
- IRE-032-427
- IRE-032-461
- IRE-032-618 (Root Cause Analysis Report)

AOP-3B, Abnormal Shutdown Cooling Conditions Unit 1, Rev 24
AOP-3B, Abnormal Shutdown Cooling Conditions Unit 1 & 2 Basis Document, Rev 20
Condensate & Feedwater system Drawings 60702SH001, 2, 3
OI-12A, Feedwater System, Rev 44
OP-5, Plant shutdown from Hot Standby to Cold Shutdown, Rev 44
AOP-J7, Loss of 120 Volt AC or 125 Volt DC Power, Rev 19
AOP-J7, Loss of 120 Volt AC or 125 Volt DC Power Basis Document U-1, Rev 11
AOP-9A, Control Evacuation and Safe S/D Due to a Severe Control Room Fire, Unit 1, Rev 14
AOP-9A, Control Evacuation and Safe S/D Due to a Severe Control Room Fire, Unit 1 Bases Document, Rev 11
June 21, 1999, BG&E Memorandum
OI-1A, Reactor Coolant System and Pump Operation Unit 1, Rev 30
Drawing 61076SH0031, Recirc Valve 1MOV659
1C09-ALM, ESFAS 12 alarm Manual (Window H-55), Rev 35
Electrical Schematic 61076SH0011D
Electrical Schematic 60617SH0011
OI-19, Instrument Air, Rev 27
Set Point File for System 019 – Compressed Air System
1C-13 ALM, SRW and MISC Station Services alarm Manual (Window K-26), Rev 53
Compressed Air System Drawing 60712SH0001
Compressed Air System Description No. -019, Rev 7
EOP-5, Loss of Coolant Accident, Technical Bases Document for Step W, Rev 23
Technical Specification Bases B 3.6.8, Rev 2
Simulator plots of feed flow, steam flow and S/G pressure following all RCPs tripping
Technical Specification 3.1.8, Special Test Exception, Modes 1 and 2, Amendment No. 201
Technical Specification 3.2.3, Total Integrated Radial Peaking Factor, Rev 3
PSTP-2, Initial Approach to Criticality and Low power Physics Testing Procedure, Rev 30
PSTP-3, Escalation to Power Test Procedure, Rev 30
EOP-5, Loss of Coolant Accident Unit 1, Rev 22
EOP-5, Loss of Coolant Accident, Unit 1 Technical Basis Document, Rev 23
Unit 1 EOP Attachments, Rev 18
Hydrogen Recombiner Technical Manual, Jan 28 1985
OI-41A, Hydrogen Recombiners, Rev 10
OI-22A, Main Exhaust Fan System, Rev 8
OI-22D, Fuel Handling Area Ventilation System, Rev 15
OI-25C, Main Exhaust Fan System
OI-36, Containment Purge System, Rev 29
FH-305, Core alterations, Rev 12
Auxiliary & Waste Processing Supply Fan 21 Drawing 63085SH0003
Waste Processing Exhaust Fan 21 Drawing 63085SH0064
Heating and Ventilation Main Plant Exhaust Fans 21 & 22 Drawing 63085-D SH11

**ATTACHMENT 4**
**UNSATISFACTORY TEST ITEMS**

Written Examination – Reactor Operator Portion

4. K/A mismatch – question replaced.
5. K/A mismatch and two implausible distractors – question replaced.
6. Two Implausible distractors – question replaced.
7. K/A mismatch – question replaced.
8. KA mismatch – question replaced.
16. K/A mismatch – question replaced.
17. Two Implausible distractors – question replaced.
18. KA mismatch – question revised.
19. K/A mismatch – question replaced.
21. Two Implausible distractors – question replaced.
22. K/A mismatch – question replaced.
23. K/A mismatch – question replaced.
24. Three implausible distractors – question revised.
25. K/A mismatch – question replaced.
27. Two correct answers – question revised.
28. Two correct answers – question revised.
33. Two correct answers – question replaced.
34. Low discriminatory value and K/A mismatch - question replaced.
39. There was no correct answer – question revised.
48. K/A mismatch – question replaced.
53. KA mismatch – question revised.
54. Three implausible distractors – question revised.
74. Two implausible distractors – question replaced.

Written Examination – Senior Reactor Operator Portion

1. K/A mismatch – question replaced and K/A resampled.
2. K/A mismatch – question replaced and K/A resampled.
3. K/A mismatch and two implausible distractors – question replaced. K/A resampled.
4. K/A mismatch – reselected K/A and kept question.
7. K/A mismatch – question replaced and K/A resampled.
11. K/A mismatch and two implausible distractors – question revised.
15. Two implausible distractors – question revised.
16. There was no correct answer – question replaced.
18. There was no correct answer – question revised.
20. Three implausible distractors – question replaced.
21. K/A mismatch – question replaced.
23. Not an SRO level question - question replaced.
24. K/A mismatch – question replaced.
25. K/A mismatch – question replaced and K/A resampled.

Operating Examination

One RO admin JPM pertaining to making an one hour report was replaced because it was determined to not be appropriate for an RO level license applicant.

One SRO admin JPM pertaining to a power ratio setpoint calculation and setpoint adjustment was replaced because is determined to be a system JPM versus an admin JPM.

Two system JPMs were replaced because one was too simplistic in that only one breaker manipulation was involved and the other JPM could not be performed in the plant because the operations department denied access to a high risk cabinet.

During the exam week two admin JPMs (1: Evaluating a containment closure deviation and 2: Performing a risk assessment of an RCA activity) were replaced because a review of the applicants responses indicated that there were two correct answers for the one JPM and the other had no correct answer.  The validation process did not uncover these flaws.

## ATTACHMENT 5
## SUMMARY OF LICENSEE POST-EXAM COMMENTS & NRC RESPONSES

| Question Number | Licensee Post-Exam Comments | NRC Responses |
|---|---|---|
| RO 5 | Accept A and B | Only B will be accepted |
| RO 13 | Accept A and B | Delete question |
| RO 25 | Delete question | Agree |
| RO 28 | Accept A and B | Agree |
| RO 31 | Accept B and D | Delete question |
| RO 40 | Accept B and C | Delete question |
| RO 54 | Delete question | Agree |
| RO 60 | Accept A and D | Only D will be accepted |
| RO 61 | Accept A and B | Delete question |
| RO 68 | Delete question | Agree |
| SRO 10 | Accept A and D | Agree |
| SRO 15 | Accept B and C | Delete question |
| SRO 24 | Delete question | Agree |

Grading based upon the original (as-administered) answer key resulted in 10/11 failures with one SROU applicant passing.

Grading based solely upon the licensees post-examination comments would have resulted in 5/11 failures with four SRO and two RO applicants passing.

Grading based upon the NRC's response to the licensee's post-examination comments resulted in 5/11 failures with four SRO and two RO applicants passing. (One of the SRO applicants who passed is different from the results of accepting all of the licensee's post examination comments.)

**ATTACHMENT 6**
**LICENSEE COMMENTS AND NRC RESPONSES**

See the following pages for the licensee's comments and the NRC responses. The format of this attachment is typically in the following order:

- The exam question followed by distractor analysis and cited references
- The licensee's explanation of the issue with inclusion of supporting material
- The licensee's re-grade request, question statistics, and cited references
- The NRC response to the licensee's re-grade request

**RO Written Examination Question #5 (ID: Q50610)**

Unit-1 in mode 5 on SDC with the RCS capable of being pressurized. The following conditions exist:

- RCS Temperature is 180°F
- RCS Pressure is 180 psia
- 11 & 12 SGFP are secured and tagged out
- Main & Auxiliary Feedwater is tagged out to 11 S/G for maintenance

A loss of both LPSI pumps occurs. Due to a malfunction, no charging pumps are available. Which of the following is the next course of actions for these conditions?

A. Feed and bleed the RCS using the HPSI pumps and pressurizer PORVs.

B. Align condensate to 12 S/G and bleed steam from 12 steam generator.

C. Align a containment spray pump to provide flow through the shutdown cooling heat exchanger.

D. Feed and bleed the RCS using the CS pumps and pressurizer PORVs.

The original answer accepted was B

**Original Answer Explanation**

In this condition, with a loss of both LPSI pumps the preferred order would be to align a Containment spray pump, followed by steaming using available S/Gs, and then followed by once through core cooling. However RCS pressure needs to be reduced to less than 170 PSIA to use the CS pumps. Since without Auxiliary Spray (No charging pumps) this is not possible, other means must be used.

A. Feed and bleed the RCS using the HPSI pumps and pressurizer PORVs. **Is incorrect** since the RCS is capable of being pressurized. This would be the last course of action of the available choices. A candidate might think that the other methods are not available since both S/Gs are not available and main feedwater is not available. A candidate might think that you need both S/Gs available for H/R.

B. Align condensate to 12 S/G and bleed steam from the 12 steam generator. **Is correct** since you have one S/G available for heat removal, and RCS pressure is too high for Containment spray pumps, and without a charging pump auxiliary spray is not available this is the next course of action.

C. Align a containment spray pump to provide flow through the shutdown cooling heat exchanger. **Is incorrect** since RCS pressure is greater than 170 PSIA, and auxiliary spray is not available to lower pressure (No charging pumps); this course of action is not correct for the conditions given. A candidate might not realize that RCS pressure is too high to use the CS pumps.

D. Feed and Bleed the RCS using the CS Pumps and pressurizer PORVs is incorrect since the RCS is capable of being pressurized. This would be the last course of action of the available choices. A candidate might think that the other methods are not available since both S/Gs are not available and main feedwater is not available, and may believe that since a CS Pump is the first alternative to a LPSI Pump, that the CS Pumps should be used first for OTCC.

**Original Reference**
The reference used to develop the test question was AOP-3B, Abnormal Shutdown Cooling Conditions, Section IV pages 22-25, and section VI pages 42-57.

**Licensee's Justification for Change**
The 3rd bulleted condition contained in the question "11 & 12 SGFPs are secured and tagged out" was confusing. It is not clear what the tagging boundaries for the feed pumps were. Two candidates concluded the SGFP bypass valve was part of the tagging boundary, making the condensate system unavailable to feed S/Gs. The feed and condensate systems are normally secured per OP-5 Section 6.3 when RCS temperature is approximately 220 degrees. For the 2008 outage, three tagouts were written which included shutting SGFP Bypass Valve, 1-FW-108, eliminating this flowpath.
The next course of action would be to feed 12 Steam Generator with Auxiliary Feedwater. This was not one of the available answer choices. With that answer not provided, the two candidates chose (answer A) the third option of cooling listed in the AOP, HPSI injection and flow through the PORVs. The remainder of the candidates stated they would depressurize the RCS using reactor vessel head vents, or by opening PORVs and then use the containment spray pumps. These actions are not supported by the AOP.
AOP-3B, Abnormal Shutdown Cooling Conditions, section VI. H page 52 (IF RCS TEMPERATURE CONTROL IS NOT ESTABLISHED, THEN ESTABLISH LONG TERM COOLING
The question was originally intended to direct the student to select the actions contained in AOP-3B section IV. D Prepare a Steam Generator for Heat Removal.

# Question #5 (ID:50610)

AOP-3B
Rev 24/Unit 1
Page 43 of 116

**VI. COMPLETE LOSS OF SDC WITH PRESSURIZATION OF THE RCS POSSIBLE**

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| D. PREPARE A SG FOR RCS HEAT REMOVAL. | |
| 1. **IF** the SGs are already in use for heat removal, **THEN PROCEED** to step E.2, Page 48. (continue) | |

This section of AOP-3B would have provided the necessary guidance to align Condensate to 12 S/G.

# Question #5 (ID:50610)

AOP-3B
Rev 24/Unit 1
Page 44 of 116

**VI. COMPLETE LOSS OF SDC WITH PRESSURIZATION OF THE RCS POSSIBLE**

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| D. (continued)<br><br>2. If necessary, inform Outage Management to restore at least one SG to a functional status by the following criteria:<br>• Feedwater flow path is available<br>• Steam flow path is available<br>• SG Level indication is available<br>• SG capable of being pressurized<br><br>3. **IF** only one SG can be made available, **THEN** isolate the SG which can **NOT** be made available:<br>a. Verify the MSIV is shut:<br>• (11 SG) 1-MS-4043-CV<br>• (12 SG) 1-MS-4048-CV<br>b. Verify the MSIV BYP valve is shut:<br>• (11 SG) 1-MS-4045-MOV<br>• (12 SG) 1-MS-4052-MOV<br>c. Verify the SG FW ISOL valve is shut:<br>• (11 SG) 1-FW-4516-MOV<br>• (12 SG) 1-FW-4517-MOV<br>d. Verify the S/G B/D valves are shut:<br>11 SG<br>• 1-BD-4010-CV<br>• 1-BD-4011-CV<br>12 SG<br>• 1-BD-4012-CV<br>• 1-BD-4013-CV<br><br>(continue) | 2.1 **IF** at least one SG can **NOT** be made available, **THEN PROCEED** to Step H, Page 53. |

**Question was intended to send the student to AOP-3B Section VI, Step D.3 to commence maintaining level using 12 Steam Generator as a heat sink.**

If the candidate believed that using 12 S/G as a heat sink was not available, then AOP-3B section VI.H would have been the next available option based on the provided choices.

**H. IF RCS TEMPERATURE CONTROL IS NOT ESTABLISHED, THEN ESTABLISH LONG TERM COOLING.**

| | |
|---|---|
| 1. Ensure **ALL** personnel are evacuated from the Containment.<br><br>2. Notify Radiation Safety Supervision of the intention to use Once-Through-Cooling.<br><br>(continue) | |

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VI. COMPLETE LOSS OF SDC WITH [partially obscured]R[partially obscured]HE RCS POSSIBLE

Section VI. H. 9 & 10 Directs opening PORVs

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| H. (continued)<br><br>9. Check open PORV BLOCK valves:<br>• 1-RC-403-MOV<br>• 1-RC-405-MOV<br><br>10. Open both PORVs:<br>a. Place the PORV OVERRIDE handswitches in MANUAL OPEN:<br>• 1-HS-1402<br>• 1-HS-1404 | |

| | |
|---|---|
| 15. **IF** a HPSI PP is available, **THEN PROCEED** to the appropriate step as follows:<br>• (13 HPSI PP) Step H.16, Page 57<br>• (11 HPSI PP) Step H.17, Page 59<br>• (12 HPSI PP) Step H.18, Page 61<br><br>Guidance to start a HPSI Pump<br><br>(continue) | 15.1 **IF** a HPSI PP is **NOT** available, **THEN PROCEED** to the appropriate step as follows:<br>• (11 LPSI PP) Step H.19, Page 63<br>• (12 LPSI PP) Step H.19.1<br>• (11 CS PP) Step H.20, Page 64<br>• (12 CS PP) Step H.20.1<br>• (CHG PP) Step H.21, Page 65 |

**From AOP-3B Bases:**

9-10. These steps provide the required actions to implement Once-Through-Cooling. Both PORV block valves are checked open and both PORVs are opened. The PORVs are solenoid-operated power relief valves that fail shut. They are in parallel to each other with an associated motor driven block valve, which are normally open. Both the PORV and its associated block valve have to be open for water to flow to the pressurizer quench tank. Both PORVs should be opened to ensure enough flow to allow for heat removal for all decay heat levels. The pressurizer quench tank is relatively small. Its internal capacity is 217 $ft^3$ (approximately 1623 gallons) with a design pressure of 100 PSIG. Because of the tank's design, when SI flow is initiated the tank's rupture disk will most likely break and the it's contents will discharge to the containment

13-21. The source of the water is based on equipment availability. This step provides the instructions for establishing flow into to the RCS. Since the Unit is most likely in an outage when on shutdown cooling, the availability of equipment will vary. This step gives instruction for the use of any HPSI, LPSI, CS or charging pump. The order of preference is HPSI, LPSI, CS and then the charging pumps. The charging pumps are the least preferable because the charging pumps may not have the capacity necessary to maintain the core cool. If the Unit has been shutdown for greater than 14 days, a charging pump can supply the flow required to maintain the core cool. But if the Unit is shutdown less than 14 days, the charging pumps probably do not have the capacity to cool the core. If the charging pumps are the only pumps available to inject water into the core, then they must be started to increase the time to boiling while restoring another pump (HPSI, LPSI, or CS) to a functional status.

The reactor core decay heat rate decreases logarithmically as fission products decay after the reactor is shutdown. By the end of the first day after shutdown from rated power operation, the decay heat rate is nearly 0.5%. This heat rate continues to decrease and is less than 0.2% by the end of the first month.

**Re-grade Request**
A and B should be accepted as correct responses based on confusion over SGFP tagging boundaries. Selection D "Feed and bleed the RCS using the CS pumps and pressurizer PORVs" should not be considered as a correct response since it is lower in the order of preference per AOP-3B. C is incorrect since RCS pressure is 180 psia and charging is not available to supply auxiliary spray to depressurize the RCS to 170 psia or less.

**Question Statistics**
Question 5 was missed by 9 of 11 students. Two students selected A, two selected B, six selected C, and one chose D. Candidates who chose C stated they either forgot the 170 PSIA Containment Spray Pump suction piping limit or they would depressurize the RCS using head/pressurizer vents of the PORVs. These actions are not supported by the AOP.

**Post Examination Review References**
AOP-3B, Abnormal Shutdown Cooling Conditions
AOP-3B, Abnormal Shutdown Cooling Conditions Technical Basis

**NRC Response**: The licensee contends that it is plausible to assume that the SGFP bypass valve is closed as part of the tagout boundary for the SGFPs in light of past tagout practices. Given this assumption, a logical possible answer to this question would be Choice A.

However, nothing in the question stated that the bypass valve was closed. Furthermore, the question indicated that main and auxiliary feedwater was tagged out to 11 S/G for maintenance. Therefore, no flow path to 11 S/G was available. The absence of a similar condition for 12 S/G, should lead applicants to conclude that 12 S/G was available. Thus, the NRC does not consider the assumption that S/G 12 was unavailable due to the SGFP bypass valve being closed to be acceptable. (There were no questions from the applicants pertaining to this question during the examination administration.)

As stated by the licensee, the assumption that 12 S/G is not available would lead the applicants to Choice A which is to perform a feed and bleed of the RCS. Using feed and bleed to cool the RCS is a very significant evolution as it would contaminate containment. Given this consequence, the lifting of the tag on the SGFP bypass valve in order to supply condensate water to the 12 S/G as a means of cooling the primary system is preferable to performing a feed and bleed operation of the RCS. This is supported by the following statement on page 4 of the AOP-3B Basis Document (emphasis added):

> "Section VI provides actions for a complete loss of SDC with pressurization of the RCS possible. Preliminary steps are performed, and, ***if possible a SG is made operable***, the RCS is allowed to heatup to develop Natural Circulation. If necessary, once-through-cooling is used to maintain RCS cooling and if SDC becomes available, then SDC is returned to service per the appropriate attachment."

AOP 3B provides procedural guidance to make a S/G available. Section VI, Complete Loss of SDC with Pressurization of the RCS Possible, Step D.2 says:

If necessary, inform Outage Management to restore at least one SG to a functional status by the following criteria:

- Feedwater flow path is available
- Steam flow path is available
- SG Level indication is available
- SG capable of being pressurized

This procedural step precedes the actions to initiate feed and bleed to the RCS.

It should also be noted that in the Original Answer Explanation section Choices A and D were considered as “the last course of action.” These choices direct action to feed and bleed the RCS. The question asked for “the next course of actions.”

**Conclusion**: The NRC does not agree with the licensee’s comment to accept both A and B as correct answers. The NRC does not agree with the assumption regarding the unavailability of S/G 12. Furthermore, procedure AOP 3B would direct the restoration of a S/G. Therefore, the original answer (B) will remain the only acceptable answer regarding the next course of action.

**<u>Reactor/Senior Operator Question 13</u>**

Unit one is operating at 100% power when the following indications are noted:
- Pressurizer pressure is 2250 PSIA
- Pressurizer level is rising
- All B/u Htrs are ON
- AFAS Loss of Power Alarm
- Actuation SYS loss of Power alarm
- RAS Actuation Sys tripped alarm
- SIAS Actuation Sys tripped alarm
- CSAS Actuation Sys tripped alarm
- 11, 12, & 13 Charging pumps are operating
- Letdown is at minimum

Based on these indications which of the following is correct?
A. 1Y01 has been lost
B. 1Y02 has been lost
C. 1Y03 has been lost
D. 1Y04 has been lost

The original answer accepted is selection A

**Original Answer Explanation**
A is correct based on the indications listed in AOP -7J. All others are not consistent with the indications of AOP 7J.

**Original References**
The reference used to develop the test question was AOP-7J Section V actions for a loss of 1Y01 pages 12-15.

**Licensee's Justification for Change**
Selection "B" is also correct. The indications listed in the stem are indications in the AOP for both the loss of 1Y01 and 1Y02. Per AOP-7J Unit-1 Rev. 19 pages 12, 13, 25 and 26, AFAS Loss of Power, Actuation SYS loss of Power, RAS Actuation Sys tripped, SIAS Actuation Sys tripped and CSAS Actuation Sys tripped alarms are received during both a loss of 1Y01 or 1Y02.

# Question #13 (ID Q50256)

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**V. 11 120 VOLT VITAL AC INSTRUMENT BUS (1Y01)**

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| A. (continued)<br><br>11. The following alarms may actuate and indications may be affected upon loss of the bus:<br><br>1C03<br>• 11 and 12 SG Channel A pressure and level indicators fail low<br>• "12 CST LVL LO" alarm<br><br>1C04<br>• "AFAS LOSS OF POWER " alarm | |

Question #13 (ID Q50256)

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**V. 11 120 VOLT VITAL AC INSTRUMENT BUS (1Y01)**

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| A.11 (continued)<br><br>1C06<br>• Loss of Channel X PZR pressure control and indication fails low<br>• Loss of Channel X PZR level control and indication fails low<br>• Channel A Total Core Cooling Flow indication, 1-PDI-101A, fails low<br>• PZR Low Range pressure indicator, 1-PI-103, goes blank<br>• PZR pressure instrument, 1-PI-102A, fails low<br>• TM/LP Trip Setpoint indication, 1-PIA-102A fails low<br>• Loss of PAM CH A FPD, 1-CRT-1C06A<br><br>1C08<br>• "ACTUATION SYS LOSS OF POWER" alarm<br>• "ACTUATION SYS" tripped alarms for SIAS, CIS, CSAS, RAS, SGIS-A, CRS, and CVCS-A<br>• "ACTUATION SYS SENSOR CH ZD TRIP" alarm | |

# Question #13 (ID Q50256)

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**VI. 12 120 VOLT VITAL AC INSTRUMENT BUS (1Y02)**

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| A. (continued)<br><br>**NOTE**<br>Loss of power to 1Y02 causes 12 CC HX SW OUT, 1-SW-5208-CV and 12A/12B SRW HX SW BYPASS VLV, 1-SW-5157-CV valves to close.<br><br>8. Restore 12 Saltwater header:<br>a. Verify 11 CC HX is in service.<br>b. Verify 12A/12B SRW HX SW OUT valve handswitches are in OPEN.<br>• 1-SW-5211-CV<br>• 1-SW-5212-CV<br><br>9. The following alarms may actuate and indications may be affected upon loss of the bus:<br>1C03<br>• 11 and 12 SG Channel B pressure and level indicators fail low<br>1C04<br>• "AFAS LOSS OF POWER " alarm | |

Question #13 (ID Q50256)

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**VI. 12 120 VOLT VITAL AC INSTRUMENT BUS (1Y02)**

| ACTIONS | ALTERNATE ACTIONS |
|---|---|
| A.9 (continued)<br>1C05<br>• Loss of PAM CH B FPD, 1-CRT-1C05B<br>1C06<br>• Loss of Channel Y PZR pressure control and indication fails low<br>• Loss of Channel Y PZR level control and indication fails low<br>• Channel B Total Core Cooling Flow indication, 1-PDI-101B, fails low<br>• PZR Low Range pressure indicator, 1-PIC-103-1, fails low<br>• PZR pressure instrument, 1-PI-102B, fails low<br>• TM/LP Trip Setpoint indication, 1-PIA-102B fails low<br>• Loss of PAM CH B FPD, 1-CRT-1C06B<br>1C08<br>• "ACTUATION SYS LOSS OF POWER" alarm<br>• "ACTUATION SYS " tripped alarms for SIAS, CIS, CSAS, RAS, SGIS-B, CRS, and CVCS-B<br>• "ACTUATION SYS SENSOR CH ZE TRIP" alarm | |

The AOP-7J Bases describes the failure responses for Reactor Coolant System Instrumentation pressurizer pressure, pressurizer level, pressurizer heater control and charging and letdown controls following a loss of 1Y01 or 1Y02. The responses for either channel are identical to the other and the operator is able to distinguish between the two but that information is not contained in the question stem. The stem does not provide information as to which pressurizer level and pressure instrument channels, X or Y is selected. There is no 'normal' lineup for these instruments that is procedurally or operationally directed.
In accordance with AOP-7J Bases SECTION NUMBER: V. 11 120 VOLT VITAL AC INSTRUMENT BUS (1YO1) the following failure responses are expected:

- 1-PIC-100X is de-energized and fails down scale. If pressure control were to remain in Channel X it would be sending a minimum pressure signal calling for the PZR heaters to be on. Channel Y is selected so that a valid PZR pressure signal is used to automatically control pressure. RRS cabinet 1C31, Channel X, is de-energized so Reactor Regulating System is switched to Channel Y.
- 1-LIC-110X is de-energized and fails down scale sending a low level signal to the PZR level control circuit resulting in all charging pumps starting and letdown reducing to minimum. Channel Y is selected so that a valid PZR level signal is used to automatically control level. Pressurizer low level cutoff is normally selected to X/Y. PZR heaters will be interlocked off due to the control channel X until the Y position is selected.

# Question #13 (ID Q50256)

**SECTION NUMBER:** V. 11 120 VOLT VITAL AC INSTRUMENT BUS (1Y01)
**BLOCK STEP:** V.A. RESPOND TO A LOSS OF 11 120 VAC INSTRUMENT BUS (1Y01).

1. 1-PIC-100X is de-energized and fails down scale. If pressure control were to remain in Channel X it would be sending a minimum pressure signal calling for the PZR heaters to be on. Channel Y is selected so that a valid PZR pressure signal is used to automatically control pressure. **[P0056]**

2. RRS cabinet 1C31, Channel X, is de-energized so Reactor Regulating System is switched to Channel Y. **[P0056]**

3. 1-LIC-110X is de-energized and fails down scale sending a low level signal to the PZR level control circuit resulting in all charging pumps starting and letdown reducing to minimum. Channel Y is selected so that a valid PZR level signal is used to automatically control level. **[P0056]**

4. Pressurizer low level cutoff is normally selected to X/Y. PZR heaters will be interlocked off due to the control channel X until the Y position is selected. **[P0056]**

In accordance with AOP-7J Bases SECTION NUMBER: VI. 12 120 VOLT VITAL AC INSTRUMENT BUS (1Y02) the following failure responses are expected:

- 1-PIC-100Y is de-energized and fails down scale. If pressure control were to remain in Channel Y it would be sending a minimum pressure signal calling for the PZR heaters to be on. Channel X is selected so that a valid PZR pressure signal is used to automatically control pressure.

- 1-HIC-110 is being given erroneous signals from PZR level program from Reactor Reg and 1-LIC-110Y, which is de-energized. So the controller is placed in manual until PZR level control signal and Reactor Reg are restored in steps 3, 4 & 6.
- RRS cabinet 1 C32, Channel Y, is de-energized so Reactor Regulating System is switched to Channel X. 1-LIC-110Y is de-energized and fails down scale sending a low-level signal to the PZR level control circuit resulting in all charging pumps starting and letdown reducing to minimum. Channel X is selected so that a valid PZR level signal is used to automatically control level.

Pressurizer low level cutoff is normally selected to X/Y. PZR heaters will be interlocked off due to the control Channel Y until the X position is selected.

# Question #13 (ID Q50256)

**SECTION NUMBER:** VI. 12 120 VOLT VITAL AC INSTRUMENT BUS (1Y02)
**BLOCK STEP:** VI.A. RESPOND TO A LOSS OF 12 120 VAC INSTRUMENT BUS (1Y02).

1. 1-PIC-100Y is de-energized and fails down scale. If pressure control were to remain in Channel Y it would be sending a minimum pressure signal calling for the PZR heaters to be on. Channel X is selected so that a valid PZR pressure signal is used to automatically control pressure. **[P0056]**
2. 1-HIC-110 is being given erroneous signals from PZR level program from Reactor Reg and 1-LIC-110Y, which is de-energized. So the controller is placed in manual until PZR level control signal and Reactor Reg are restored in steps 3, 4 & 6.
3. RRS cabinet 1C32, Channel Y, is de-energized so Reactor Regulating System is switched to Channel X. **[P0056]**
4. 1-LIC-110Y is de-energized and fails down scale sending a low-level signal to the PZR level control circuit resulting in all charging pumps starting and letdown reducing to minimum. Channel X is selected so that a valid PZR level signal is used to automatically control level. **[P0056]**
5. Pressurizer low level cutoff is normally selected to X/Y. PZR heaters will be interlocked off due to the control Channel Y until the X position is selected. **[P0056]**

**Re-grade Request**

Selections “A” and “B” are correct based upon missing information in the stem stating the positions of the pressurizer pressure and pressurizer level control channel selector switches. Without a specific set of handswitch positions identified, either answer is correct.

**Question Statistics**
Question 13 was missed by 9 of 11 students. Two students selected the original correct answer and nine students selected B.

**Post Examination Review References**
AOP-7J, Loss of 120 Volt AC or 125 Volt DC Power, Revision 19.
AOP-7J, Loss of 120 Volt AC or 125 Volt DC Power Basis Document U-1, Rev 11.

**NOTE:** The stem of the question also contains an error. The Pressurizer backup heaters would not be on since a Pressurizer level instrument LIC 110X or 110Y would fail low causing the low level heater cutout. The normal position of the PZR HTR LO LVL CUT-OFF SEL switch is X/Y, therefore, with either 1Y01 or 1Y02 de-energized, backup heaters would be interlocked off. This error is inconsequential to candidates' choice of answers.

**NRC Response**: The licensee contends that because there is no "normal" system alignment, the loss of either 1Y01 or 1Y02 would cause the symptoms provided in the question and therefore either Choice A or B should be accepted as correct answers.

While reviewing this question prior to administration, the examiners challenged the examination author regarding the "normal" system line up. The examiners were assured that the applicants would know what the "normal" system line up is and that the question would elicit only one correct response. However, this post-examination comment by the licensee refuted the examination author's contention of a known "normal" line up. (There were no questions from the applicants pertaining to this question during the examination administration.)

During the post-examination review of the information provided by the licensee for this question, the NRC noted that in order for either Choice A or B to be correct, all of the pressurizer controls (pressure, level, and low level heater cutoff) would have to be selected to either the X or Y channel at the same time. Although there is no specified "normal" line up as pressurizer pressure or level controls could be selected to either the X or Y channel, the pressurizer level heater cutoff controller is normally selected to the X/Y position as per AOP-7J Basis Document for Section IV, Block Step V.A.4 and Section VI, Block Step VI.A.5. In this X/Y position, a loss of either power supply would **not** result in pressurizer back up heaters coming on (as mentioned by the licensee above). The licensee contends that this is inconsequential to the applicants' choice of answers. The NRC disagrees because the question was technically incorrect. The question stem provided a set of conditions that were not consistent with any of the causes listed as choices. Thus, for the conditions provided in the question, there is no correct answer.

**Conclusion**: The NRC does not agree with the licensee's comment to accept either Choice A or B as correct answers. Given that the information provided in the question did not accurately reflect operating practices at Calvert Cliffs and therefore would normally not occur, there is no correct answer. This question will be deleted from the examination.

**<u>Reactor/Senior Operator Question 25</u>**

A fire in the Unit1 Cable Spreading room has occurred. The SM has determined that a Control Room evacuation is necessary and AOP-9A should be implemented. Which of the following sets of actions are required to be completed within the first 30 minutes of CR Evacuation to prevent damage to plant equipment?

A. Trip the RCPs AND start the 0C Diesel Generator
B. Start the 0C Diesel Generator AND Establish Charging flow
C. Trip the RCPs AND Trip MCC-104 load center
D. Establish AFW flow AND Establish Charging flow
The original answer accepted was A

**Original Answer Explanation**

1. Trip the RCPs AND start the 0C Diesel Generator - correct per AOP9A basis IV.C and notes III C. 2
2. Start the 0C Diesel Generator AND Establish Charging flow - Not correct, Charging flow not required until 60 minutes
3. Trip the RCPs AND Trip MCC-104 load center - Not correct, trip MCC-104 load center does not have a time limit
4. Establish AFW flow AND Trip MCC-104 load center - Not correct , charging flow not required for 60 minutes

**Original References**

AOP-9A Basis page 1
AOP-9A page 5

**Licensee's Justification for Change**

There is no correct answer. The question states: Which of the following sets of actions are required to be completed within the first 30 minutes of Control Room evacuation **to prevent damage to plant equipment?** No set of answers presented are based upon preventing equipment damage. Starting the OC DG within 30 minutes is a time saving step per the AOP-9A Basis document Section IV Step C.

# Question #25 (ID Q50475)

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III. (continued)

C. **NOTES**

1. Offsite power may be lost at any time during this event. Evolutions are performed to place 0C Diesel Generator on 4KV Bus 24 and 4KV Bus 11 and to secure the remaining 4KV Busses. In the event a loss of offsite power occurs within the first hour, a Station Blackout will be in effect until the 0C Diesel Generator and necessary support equipment are re-aligned.

2. The 0C DG must be started within 30 minutes of a loss of power to 07 4KV Bus to prevent the need for prelubing the 0C DG using the pneumatic prelube pumps.

3. It is important to establish Auxiliary Feedwater flow within 30 minutes and Charging flow within 60 minutes.

4. During the post-fire shutdown, the Reactor Coolant System parameters shall be maintained within those predicted for a loss of offsite power. The fission product boundary integrity shall not be affected. The Reactor Coolant makeup function shall be capable of maintaining level indication in the Pressurizer. No other plant accidents are assumed to occur except as precipitated by the fire.

# Question #25 (ID Q50475)

AH. **(OSO)** TAKE LOCAL CONTROL AND START 0C DIESEL GENERATOR

1. **IF** 0C Diesel Generator is **NOT** running
**AND** the 07 4KV Bus has been de-energized for 30 minutes or longer,
**THEN** initiate pneumatic prelube as follows:

   a. Open 0C1 PNEUMATIC PRELUBE PUMP SUCTION VALVE, 0C1-DLO-2 (at 0C1 pneumatic prelube pump).

   b. Open 0C1 PNEUMATIC PRELUBE PUMP DISCHARGE VALVE, 0C1-DLO-14 (under 0C1 Aux Desk).

   c. Open 0C2 PNEUMATIC PRELUBE PUMP SUCTION VALVE, 0C2-DLO-2 (at 0C2 pneumatic prelube pump).

   d. Open 0C2 PNEUMATIC PRELUBE PUMP DISCHARGE VALVE, 0C2-DLO-14 (under 0C2 Aux Desk).

   e. Open one 0C Pneumatic Prelube Pump Air Bottle outlet valve (35' elevation).

   f. Adjust 0C PNEUMATIC PRELUBE PUMP AIR REGULATOR, 0-DLO-10180-PCV, to a maximum of 110 PSIG on 0C PNEUMATIC PRELUBE PUMP AIR SUPPLY REGULATOR OUTLET PI, 0-PI-10180, to start the Prelube Pump turning.

(continue)

# Question #25 (ID Q50475)

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**AOP-9A**
**Unit 1**
**BASES DOCUMENT**

IV. ACTIONS

A. This step describes the affected area, defines a severe fire and gives the Shift Manager guidance in determining if the fire is a severe fire.

B. Manual Reactor TRIP is initiated to promptly change plant modes. Manual Reactor TRIP is an allowable action taken per Generic Letter 86-10, "Questions and Answers," Section 3.8.4.

C. Credit is not taken for Control Room initiated trips of the Main Turbine, SGFPs, and shutting the MSIVs (i.e., an exemption was not requested), but are considered prudent actions to minimize the probability of an over-cooling scenario. If the 0C DG can be emergency started here it will save the time of pneumatically prelubing the DG prior to start. If it can't be started here it will be started later in the procedure. Upon loss of component cooling, the RCPs have been evaluated to operate 20 minutes without seal failure. Since the wiring in the Control Room has not been evaluated, the RCPs must be tripped locally within 20 minutes. It is considered a prudent action to trip the RCPs from the Control Room prior to leaving. (Reference: WCAP-16175-P, Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants) 1100

# Question #25 (ID Q50475)

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AH. The OSO will take local control of and start the 0C DG. The 0C DG will be used to power both the 11 and 24 4KV Buses simultaneously. If the 0C DG was not started prior to leaving the Control Room and the 07 4KV Bus has been de-energized for 30 minutes or longer then the pneumatic prelube must be performed. [B0255]

# Question #25 (ID Q50475)

BGE

MEMORANDUM

Auxiliary Systems Engineering Unit
G:\PES\990621-200.DOC

TO: R. J. Martin

FROM: S. J. Loeper

DATE: June 21, 1999

SUBJECT: Prelube time for SACM engines

Due to the required operator actions during plant responses involving the SACM engines, PES requested clarification of the pre-lubrication requirements for the SACM engines. SACM has confirmed that a limited number of engine starts without prelube (up to a maximum of 30 minutes) is acceptable. Therefore, to reduce the time delay for OC DG connection to a 4 KV bus and reduce the required operator actions during a Station Blackout, PES considers it acceptable to start the OC DG without prelube for up to 30 minutes without requiring pneumatic prelube. Starting of the OC DG without prelube for up to 30 minutes should only be utilized for loss of off site power scenarios.

If you have any questions, please contact Steve Loeper at x4734.

**Re-grade Request**
Question #25 should be deleted due to no correct answer. The basis for starting the OC DG within 30 minutes is to save time during a loss of power event.

**Question Statistics**
Question 25 was missed by 7 of 11 students. Four students selected A, five selected C and two selected D. Candidates stated they knew that tripping RCPs had to be completed within 20 minutes, initiating Auxiliary Feedwater had to be completed within 30 minutes and charging initiated with 60 minutes.

**Post Examination Review References**
AOP-9A Unit 1, rev. 11
AOP 9A Bases
Memo from Steve Loeper, system engineer to Bob Martin, on shift SRO

**NRC Response**: The license contends that there is no correct answer because there is no choice in which both of the actions that are listed would be required "to prevent damage to plant equipment." The intended correct answer was Choice A: Trip the RCPs AND start the 0C Diesel Generator. Based upon the information provided above, the NRC agrees that the RCPs need to be tripped within 20 minutes to prevent damage to the RCP seals. The NRC also agrees with the licensee that starting the 0C Diesel Generator within 30 minutes is only done for efficiency so that pneumatic prelubing will not need to be done. The question asked for a set of conditions that would prevent damage to plant equipment. Starting the 0C Diesel Generator within 30 minutes is not an action that is taken to prevent damage to plant equipment. Therefore Choice A is not correct.

**Conclusion**: The NRC agrees with the licensee's recommendation to delete the question because there is no correct answer provided among the choices. The question will be deleted.

**Reactor/Senior Operator Question 28**

Using the provided Reference and given the following conditions on Unit-2.

- Pressurizer Pressure = 315 PSIA
- RCS Tcold = 140F
- S/G Temperature = 90F
- Pressurizer Level = 160 inches
- 4KV Bus Voltage = 4130KV
- 13.8 KV Bus Voltage = 14.2 KV

Which of the following conditions would prevent starting 21A RCP per plant operating procedures?

A. A pressurizer level control malfunction causes pressurizer level to rise to 172 inches and stabilizes
B. A heat up causes RCS Temperature to rise to 155F and stabilizes
C. A voltage regulator perturbation causes 4KV bus voltage to lower to 4110 KV
D. An electrical perturbation causes 13 KV bus voltage to rise to 14.8 KV and stabilizes.

Original answer accepted was A

**Original Answer Explanation**
Per OI-1A Section 6.1.B starting requirements for an RCP, S/G temperature no more that 60F below RCS temperature, pressurizer level less than 170 inches, RCS pressure and temperature within the limits of figure 17, 4KV bus voltage greater than 4100 volts and 13.8KV bus voltage less than or equal to 14.8 KV.

A. A pressurizer level control malfunction causes pressurizer level to rise to 172 inches and stabilizes is correct since pressurizer level has to be less than 170 inches.
B. A heat up causes RCS temperature to rise to 155F and stabilizes is incorrect since RCS temperature has to be less that 60F above S/G temperature 90+ 60 =150. 155F is 55F less than RCS temperature and still within limits. **[Error 155-90=65, > limit]**
C. A voltage regulator perturbation causes 4KV bus voltage to lower to 4110 volts and stabilizes is incorrect since bus voltage is greater than 4100 volts.
D. An electrical perturbation causes 13KV bus voltage to rise to 14.8 KV and stabilizes is incorrect since the limit is less than or equal to 14.8 KV.

**Original References**
OI-1A, Reactor Coolant System, pages 5-13
OI-1A, Reactor Coolant System figure 17

**Licensee's Justification for Change**
The following is justification why selection B is also correct based upon the initial provided data and plant condition change. Selection B states the following: A heat up causes RCS temperature to rise to 155F and stabilizes.
OI-1A stipulates that the first RCP shall not be started with one or more Unit 2 cold leg temperatures less than 301F unless:

- The pressurizer water level is less than 170 inches
- For Unit 2, Steam Generator temperature limits are as follows
    - S/G temperature greater than 30F above RCS temperature
    - Steam Generator temperature is no more than 60F below RCS temperature
- Unit 2 pressurizer pressure is between 250 and 320 PSIA as indicated on PZR LO RANGE PRESS 2-PI-103 or 2-PI-103-1

**5.0 PRECAUTIONS (Continued)**

H. The first RCP shall **NOT** be started with one **OR** more Unit 1 RCS cold leg temperatures less than or equal to 365° F **OR** Unit 2 RCS cold leg temperature less than or equal to 301° F unless the following conditions exist: **[B0064]**

> **NOTE**
> - Pressurizer level must be verified using temperature compensated level indication, LI-103, **AND** computer point L110X! or L110Y!.
> - LI-103 and L110X! **OR** LI-103 and L110Y! shall agree within (±)13 inches, unless otherwise evaluated as acceptable by engineering.

- The Pressurizer water volume is less than 170 inches

**5.0.H** **PRECAUTIONS** **(Continued)**

**NOTE**

- If SDC is secured, RCS temperature shall be determined using RCS $T_{AVG}$ (the average of RCS $T_{HOT}$ and $T_{COLD}$) from the loop with the largest delta T
  - (11(21) loop) 1(2)-TI-112H, 1(2)-TI-112C
  - (12(22) loop) 1(2)-TI-122H, 1(2)-TI-122C
- If SDC is in operation, RCS temperature shall be determined using the SDC Temperature Recorder, 1(2)-TR-351, **OR** computer points as follows:
  - If **NO** LPSI flow rate changes have been made in the previous 30 minutes, using the average SDC Temperature (the average of FROM RCS(red pen) and TO RCS(blue pen) **OR** computer points T351X and T351Y
- **IF** SDC is in operation, **AND** RCS temperature can **NOT** be obtained by averaging, **THEN** RCS temperature shall be determined using the SDC Temperature Recorder, 1(2)-TR-351, **OR** computer points as follows:
  - S/G 30° F above RCS temperature: RCS temperature shall be determined using TO RCS(blue pen) **OR** computer point T351X
  - S/G below RCS temperature: RCS temperature shall be determined using FROM RCS(red pen) **OR** computer point T351Y
- Steam Generator temperature is obtained locally using a hand-held surface instrument on the Steam Generator shell between the Steam Generator Tube Sheet and water level covering the tubes

- For Unit 1, Steam Generator temperature limits are as follows:
  - S/G temperature is no greater than 30° F above RCS temperature **[B0064]**
  - **IF** RCS temperature is 146° F or less, **THEN** S/G temperature is no more than 50° F below RCS temperature **[B0410]**
  - **IF** RCS temperature is greater than 146° F, **THEN** S/G temperature is no more than 60° F below RCS temperature **[B0410]**
- For Unit 2, Steam Generator temperature limits are as follows:
  - S/G temperature is no greater than 30° F above RCS temperature **[B0064]**
  - S/G temperature is no more than 60° F below RCS temperature **[B0410]**

The initial steam generator temperature was 90°F and a heat up caused RCS temperature to rise to 155°F. The difference between 155°F and 90°F (155-90= 65) equals 65°F which exceeds the RCP start limit of S/G temperature of no more than 60°F below RCS temperature.

**Re-grade Request**

Response B is also correct. A math error on the original answer justification incorrectly eliminated B as a correct choice. Accept A and B.

**Question Statistics**

Question 28 was missed by 9 of 11 students. Two candidates selected A, two selected B, three candidates chose C and four selected D.

**Post Examination Review References**
OI-1A, Reactor Coolant System

**NRC Response**: The licensee contends that there are two correct answers to this question. The question asked the applicants to choose from a list of conditions that would prevent starting an RCP. Choice A was the intended correct answer (and still is a correct response). However, due to an oversight in the examination review process, the condition stated in Choice B was worded such that this was also a correct answer. An RCP cannot be started if the RCS is greater than 60 °F above the S/G temperature. The licensee's comment correctly reveals that the temperature difference between the RCS and the S/G as presented in the question was 65°F.

**Conclusion**: The NRC agrees with the licensee's recommendation. There are two conditions listed among the choices that would prevent starting an RCP. Choices A or B will be accepted as correct answers to this question.

**Reactor/Senior Operator Question 31**

Which of the following is the most likely reason for this condition?
"SI PPS RECIRC MOV 659 CLOSED RAS BLOCKED" Alarm is ON

A. MINI FLOW RETURN TO RWT ISOL, 1- SI-659 MOV, is shut with an inadvertent RAS present

B. MINI FLOW RETURN TO RWT ISOL MOV, 1- SI-659 MOV is shut with no RAS present

C. SI PP RECIR LOCKOUT handswitch, 1-HS-3659A, is ON and RAS present

D. MINI FLOW RETURN TO RWT ISOL, 1-SI-659-MOV shut and SI PP RECIR LOCKOUT handswitch, 1-HS-3659A in ON

Original answer accepted was B

**Original Answer Explanation**
Per Alarm Manual for 1C09 window H-55 Different sets of conditions will give the alarm.

B. MINI FLOW RETURN TO RWT ISOL MOV, 1- SI-659 MOV is shut with no RAS present will give this alarm

A, C, D have conditions that do not fully satisfy any of the three requirements to get the alarm

**Original References**
Alarm manual 1C09 window H-55 page 86.

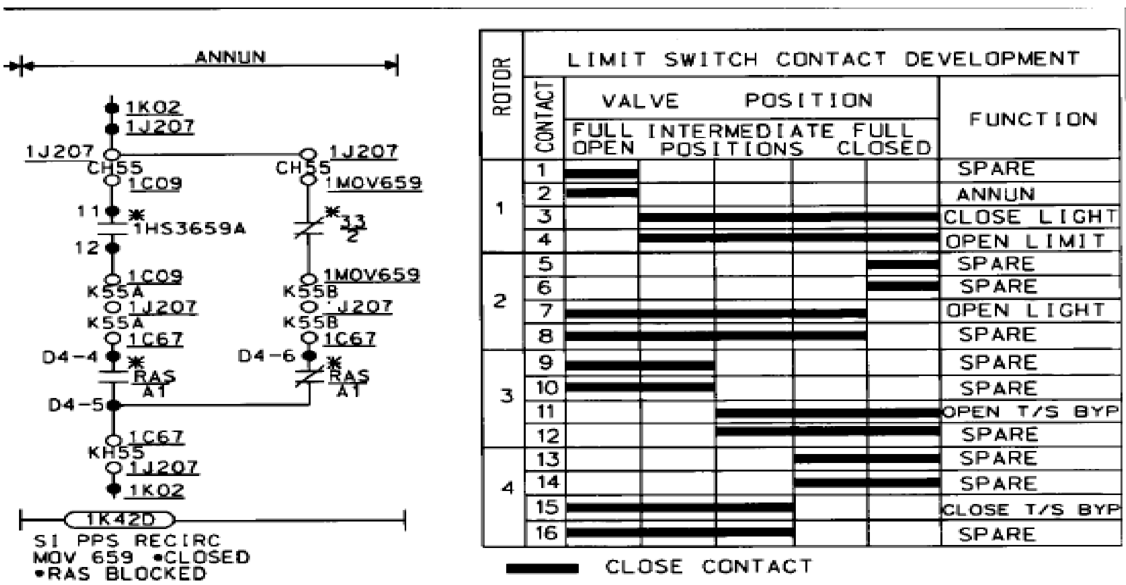
**Licensee's Justification for Change**
The following is the justification for changing the answer key to include selection D as a correct response in addition to selection B. Selection D states MINI FLOW RETURN TO RWT ISOL, 1-SI-659-MOV shut and SI PP RECIRC LOCKOUT handswitch, 1-HS-3659A in ON.
Selection "D" will also cause the "SI PPS RECIRC MOV 659 CLOSED RAS BLOCKED" alarm to actuate with MOV-659 shut and 1-HS-3659A in ON. The alarm circuitry logic seeks a combination of MOV 659 position, RAS actuation and 1-HS-3659A position.

| DEVICE | SETPOINT | WINDOW | H-55 |
|---|---|---|---|
| 1-SI-659-MOV<br>1-HS-3659A<br>RAS relay | N/A | SI PPS RECIRC<br>MOV 659 • CLOSED<br>• RAS BLOCKED | |

**POSSIBLE CAUSES**

- SI PP RECIRC LOCKOUT handswitch, 1-HS-3659A, in LOCKOUT with a RAS signal present
- MINI FLOW RETURN TO RWT ISOL MOV 659, 1-SI-659-MOV, shut without RAS signal present
- MINI FLOW RETURN TO RWT ISOL MOV 659, 1-SI-659-MOV, shut **AND** SI PP RECIRC LOCKOUT handswitch, 1-HS-3659A, in LOCKOUT with or without an RAS signal present

Referencing electrical print 61076SH0031 (REACTOR SAFEGAURDS C.S. & S.I. PUMPS RECIRC VALVE 1MOV659) when 1MOV659 is shut without a corresponding RAS, contact 33/2 will open and interrupt power to the alarm circuitry and actuate the alarm.




Under normal configuration, 1HS3659A is in the lockout position which prevents aligning power to reposition 1MOV659. When handswitch 1HS3659A is placed in the ON position, power is aligned to 1MOV659 and the MOV is able to be repositioned by an operator. If 1HS3659A is in the ON position and an operator shuts 1MOV659 without a RAS present, contact 33/2 will open and the “SI PPS RECIRC MOV 659 CLOSED RAS BLOCKED” alarm will actuate. The electrical print shows that if 1HS3659A is placed in the ON position, contact 7 and 8 will shut and align power to 1MOV659 open/close circuitry and allow repositioning.

Based upon electrical print 61076SH0031, 1MOV659 can be shut with 1HS3659A in the ON position and satisfy the circuitry to actuate the “SI PPS RECIRC MOV 659 CLOSED RAS BLOCKED” alarm. The function of 1-HS-3659A is to supply power for positioning 1MOV-659 such as in EOP-5 when preparing for RAS actuation.

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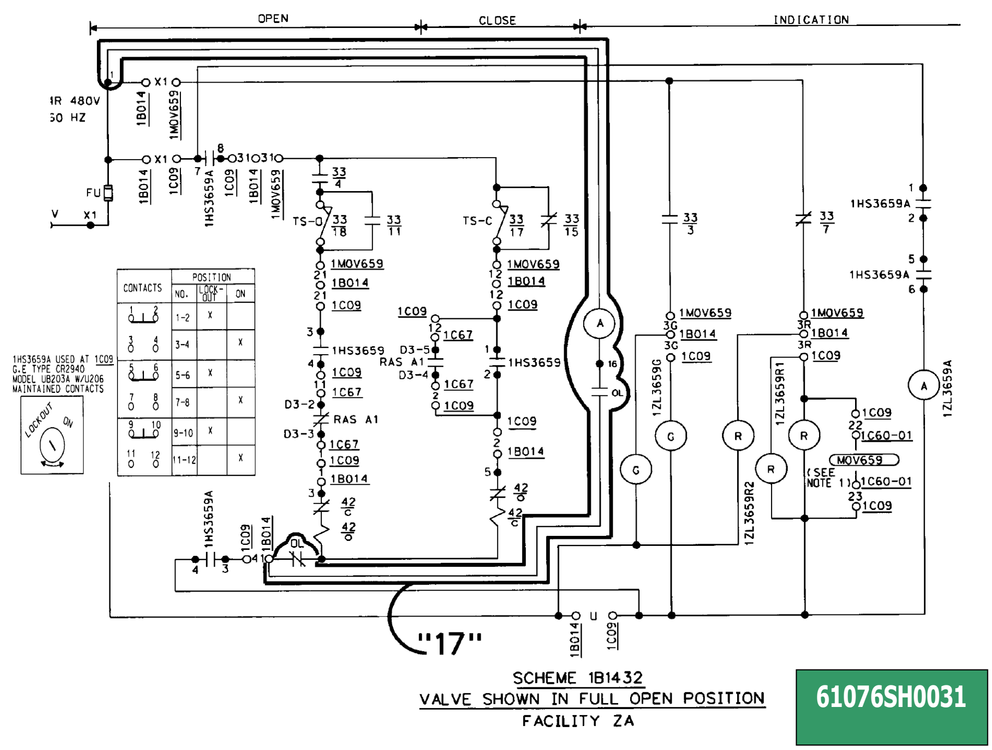
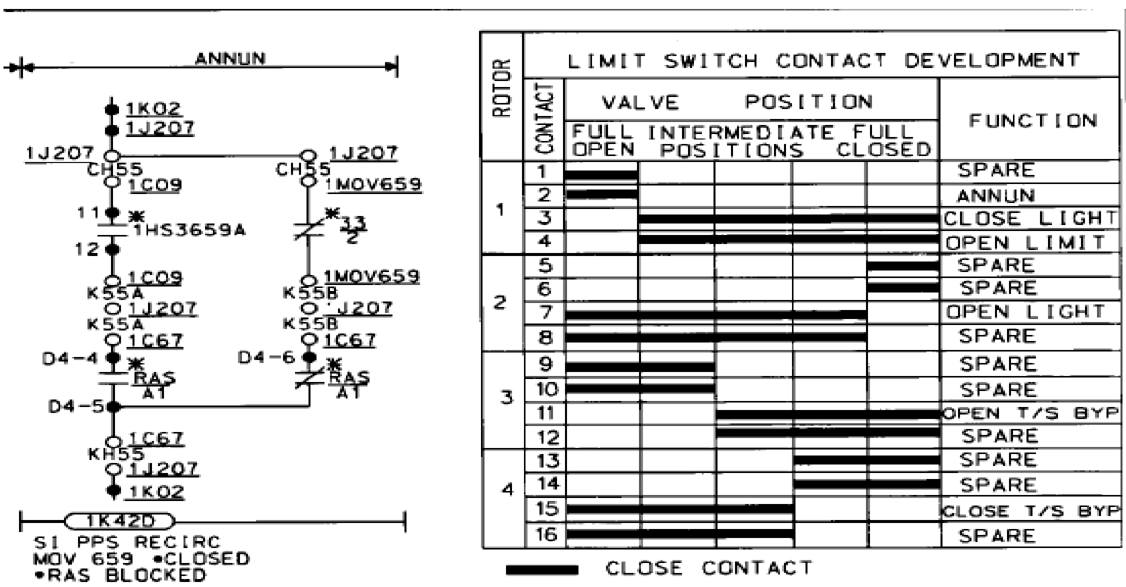



**Re-grade Request**
There are two correct answers. B and D should be accepted as correct responses for this question based on the electrical prints and logic diagrams.

**Question Statistics**
Question 31 was missed by 8 of 11 students. Three candidates selected B, four candidates selected C and four candidates selected D.

**Post Examination Review References**
61076SH0031, RECIRC VALVE 1MOV659
Alarm Manual 1C09 Window H-55

**NRC Response:** The licensee contends that Choices B or D should be accepted as correct responses. The licensee's discussion above supports that Choice D is an alternate correct response. The conditions present in Choice D are 1) SI PP RECIRC LOCKOUT Handswitch 1HS-3659A in ON, 2) NO Recirculation Actuation Signal (RAS) present, and 3) 1MOV659 is CLOSED. The circuitry for the annunicator alarm (from print 61076SH0031) is shown with RAS contacts in the configuration for no RAS present. (See below.) When no RAS is present (there is no basis for the applicant to assume RAS actuation because the stem), the left branch of the alarm circuit is interrupted by the open RAS contact (from D4-4 to D4-5). The position of the in-series Lockout Handswitch contact (1HS3659A, 11-12) is immaterial as the alarm state is satisfied for the left branch when the circuit is interrupted by the absence of a RAS signal. If the MOV is closed in the no-RAS condition, the associated MOV position contact (33/2) will open, interrupting the right branch of the alarm circuit. With both circuit branches interrupted, the alarm will actuate thereby making Choice D also correct.




However, further review of this circuit logic shows that Choice A is also a correct answer. Choice A establishes conditions of 1MOV659 being closed with an inadvertent RAS present. The MOV position interrupts the right side of the alarm circuit by opening position contact 33/2. The RAS present closes a contact in the left branch (D4-4 to D4-5). The choice does not mention lockout switch position. For this choice, there is no basis for the applicant to assume other than the normal LOCKOUT position for the lockout switch since no information is provided regarding the switch. With the lockout handswitch in LOCKOUT, the left branch is also interrupted at the 11-12 contact. Likewise, with both branches interrupted, the alarm actuates.

**Conclusion:** The NRC disagrees with the licensee's recommendation to accept either Choice B or D as correct answers. Examination Standard ES-403 D.1.c states that "If three or more answers could be considered correct or there is no correct answer, the question shall be deleted." Because there are three correct choices (A, B, and D) the question will be deleted.
**Reactor/Senior Operator Question 40**

A SIAS has occurred on Unit 1. Which of the following is a correct statement for CAC operation?
A. The CACs can be started in Fast Speed at 1C09 AND at the load contactor panel.
B. The CACs can be shifted to Fast Speed at the load contactor panel ONLY
C. The CACs can be stopped from the load contactor panel ONLY
D. The CACs can be stopped at 1C09 and at the load contactor panel.

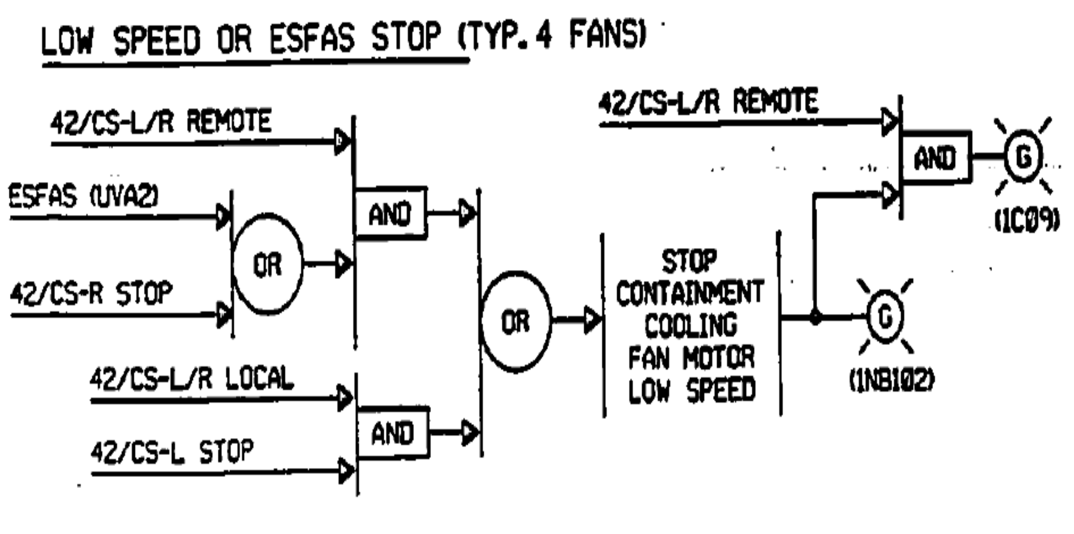
Original answer accepted was B

**Original Answer Explanation**
The CACs can be shifted to Fast Speed at the load contactor panel ONLY.
A is incorrect, with a SIAS present CACs can not be started in fast speed from the control room per LD 76 sheet 1.
B is correct since CACs can be shifted to Fast Speed at the load contactor panel ONLY.
C is incorrect, SIAS seals in per LD 76 sheet 11.
D is incorrect SIAS signal seals in per LD 76 sheet 11.

All of the answers require the candidate to be familiar with the logic sheets and/or control drawings for the CACs. If a candidate does not know the logic he could have the misconception that the CACs can be shifted to fast or stopped with a SIAS present since the H/S at 1C09 have a pull to lock feature to start them in slow. He could confuse this with the ability to pull to lock and stop the CACs. Some pumps (CCW, SW, SRW) can be pulled to lock and will not start on SIAS. This is not true for the CAC. CACs are manually started in fast speed from 1C09 when containment environment is degraded.

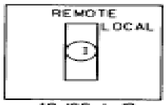
**Original References**
LD-76 Sheet 11
AOP-9A page 53

**Licensee's Justification for Change**
Selection C is also a correct response. The following is the justification for changing the examination answer key to include selection C as a correct response in addition to selection B.
Question #40 states the following: A SIAS has occurred on Unit 1. Which of the following is a correct statement for Containment Air Cooler (CAC) operation?
Selection C states the following: The CACs can be stopped from the load contactor panel ONLY. If the CAC local remote handswitch is positioned to the LOCAL position and the CAC local control handswitch is selected to STOP, the CAC will stop regardless of the SIAS actuation signal.
The CAC logic diagram shows that the SIAS signal is not sealed in, preventing securing the CAC, if the remote/local handswitch is in the REMOTE position.
If the remote/local selector switch is in LOCAL and the local control handswitch is selected to STOP, the CAC will stop.




Electrical schematic 61076SH0011D shows that a CAC can be stopped from its local load contactor if its local/remote switch is placed in LOCAL and its local control switch is placed in STOP then spring returned to NORMAL. In this configuration the CAC can be shifted to Fast or Slow Speed locally in addition to being secured during an active SIAS.

**1HS5300A1**




42/CS-L/R

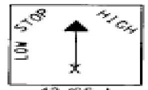
MAINTAINED CONTACTS.
KEY OPERATED.
KEY REMOVABLE IN
REMOTE POSITION

| DECK | CONTACTS | POSITION: REMOTE | POSITION: LOCAL |
|---|---|---|---|
| 1 | 1–2 | X | |
| | 3–2 | | X |
| | 5–6 | X | |
| | 7–6 | | X |

ONE DECK SHOWN
TYPICAL FOR TEN DECKS.
EXCEPT FOR TERMINAL NUMBERS

TERMINAL NUMBER = 1 1
DECK NO.
CONTACT NO.

**1HS5300A**

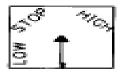



42/CS-L

SPRING RETURN TO NORMAL
FROM STOP & HIGH.
PULL TO LOCK AT LOW.
PISTOL-GRIP HANDLE

| CONTACTS | LOW | STOP | NORM | HIGH |
|---|---|---|---|---|
| 1–2 | | | | X |
| 3–4 | | | X | X |
| 5–6 | X | | | |
| 7–8 | X | | | |

**1HS5300** USED AT 1C09



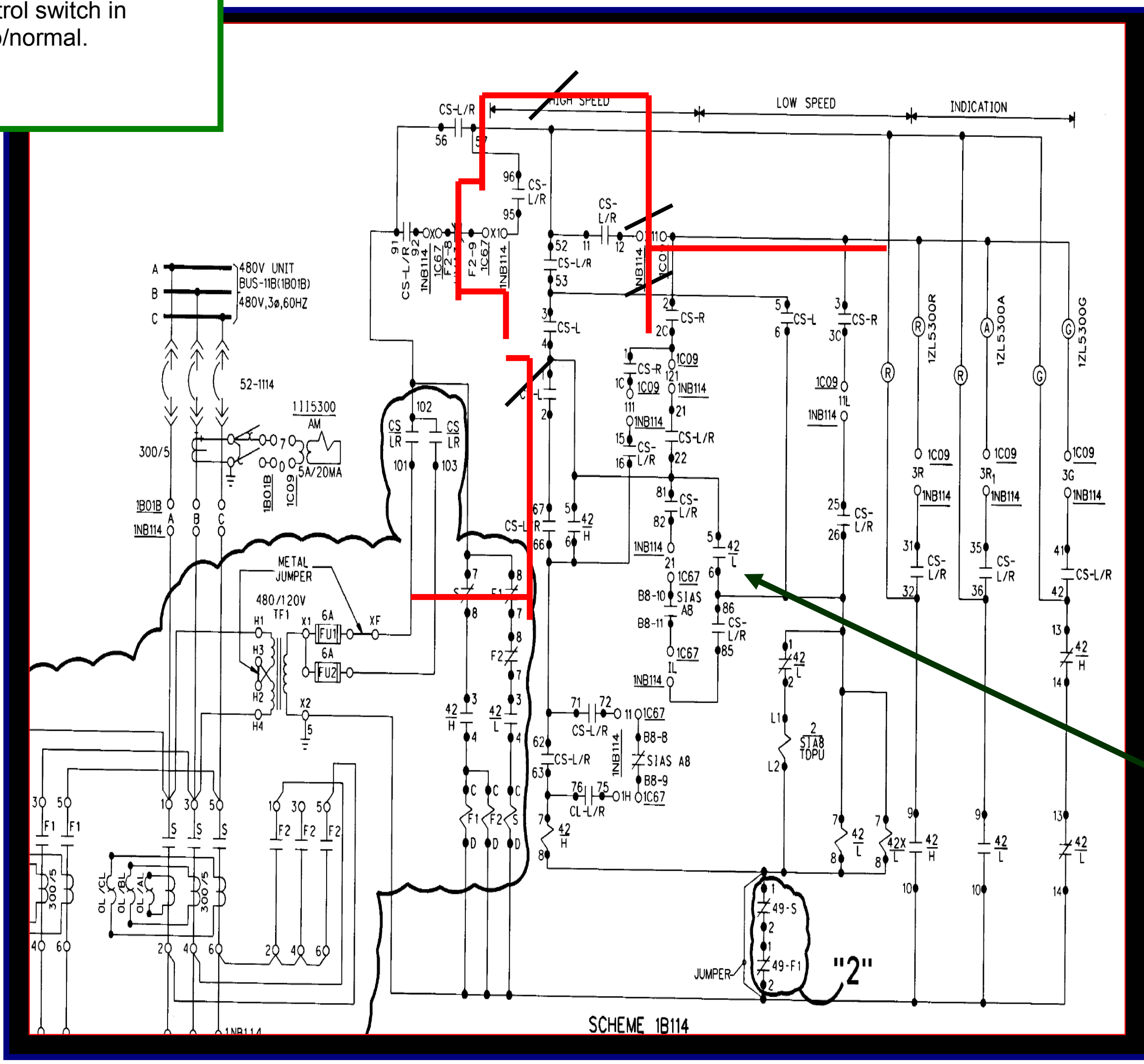

42/CS-R

| CONTACTS (HANDLE – END) | NO. | HIGH | NORMAL | STOP | LOW |
|---|---|---|---|---|---|
| 1 – 1C | 1 | X | | | |
| 2C – 2 | 2 | X | X | | |
| 3 – 3C | 3 | | | | X |
| 4C – 4 | 4 | | | | X |

SPRING RETURN TO NORMAL FROM
STOP & HIGH, PULL TO LOCK AT LOW

61076SH0011D

The trace with the red lines display the control circuit's contact alignment with the local/remote switch in local and the local control switch in stop/normal.




In this control circuit configuration, power is not supplied to the SIAS A8 contact if shut.

**Re-grade Request**
C and B should be accepted as correct answers.
A CAC can be stopped or shifted to fast speed from its local load contactor panel with a SIAS signal present.

**Question Statistics**
Question 40 was missed by 9 of 11 students. Two candidates selected B, five candidates selected C and four selected D.

**Post Examination Review References**
60617SH0011
61076SH0011D

**<u>NRC Response:</u>** The licensee contends that either Choices B or C should be accepted as correct answers. The licensee proposes accepting Choice C as a second correct answer because, contrary to the original question justification, the fan can be stopped from the local load contactor panel with a SIAS signal present. It is correct that the fan can be stopped at the local panel with a SIAS present. However, the control circuit shows that the fan can also be stopped from the control room at 1C09, by holding the spring return fan handswitch at 1C09 in the STOP position. With the 1C09 handswitch held in the STOP position the CS-R contacts (which are labeled 2-2C and 3-3C) remain open in the high speed, the SIAS low speed, and the non-SIAS low speed branches of the fan control circuit. These open contacts interrupt power to both the high (42/H) and the low (42/L) speed contactors. The fan motor line contacts will remain open when the main contactors are de-energized and the fan will be stopped. It should also be noted that Choice C states “The CACs can be stopped from the load contactor panel **ONLY**.” Therefore, Choice C is not technically correct because the fans can be stopped from the control room and at the load contactor panel. Because the CACs can be stopped at both locations, Choice D (The CACs can be stopped at 1C09 and at the load contactor panel) is technically correct. Thus, the NRC has determined in the purest technical sense that there are two correct choices: B and D.

The licensee’s contention that Choice C is correct is predicated upon an argument of reasonableness. It is not the licensee’s expectation that the CACs should be controlled (stopped) by designating an operator to hold a spring return hand switch in a position abnormal to its designed response or function. If one acknowledges this action to be unreasonable for a competent operator (and the NRC acknowledges the licensee’s perspective), then Choice C is operationally correct. This perspective supports the licensee’s contention of two correct choices: B or C.

The NRC and the licensee agree that there are two correct answers to this question; however, the agreement is not regarding the same two choices depending upon if one interprets the word “can” (which appears in all of the distractors) to mean that something is physically possible or if means that an action is in accordance with standard practice or procedure. When considered collectively (the technically correct answers and the operationally correct answers), there are three possible correct choices: B, C, and D.

Furthermore, it should be noted that it took several reviewers many hours to understand the detailed circuit logic regarding this question to come to a point to adequately assess the distractors. For an applicant to answer this question it would require either the memorization of, or the use of, the circuit logic diagrams. This makes the level of difficulty of this question to be excessive and therefore inappropriate to be on the examination. The question was originally deemed to be of an acceptable level of difficulty because the examination developer and NRC reviewers incorrectly assumed that the SIAS signal sealed in and therefore prevented the CACs from being stopped thereby eliminated Choices C and D.

**<u>Conclusion:</u>** The NRC does not agree with the licensee’s recommendation to accept either Choices B or C because there appear to be three correct answers. Examination Standard ES-403 D.1.c states that “If three or more answers could be considered correct or there is no correct answer, the question shall be deleted.” Because there can be three “correct” choices (B, C, and D) the question will be deleted.

**Reactor/Senior Operator Question 54**

Unit -1 is operating at 100% power when Instrument Air System pressure decreases to 96 psig.

Which of the following is correct?

A. Loss of Power to an IA dryer has occurred.

B. Standby Air Compressor has picked up

C. Plant Air to I/A X-Conn, 1-IA-2061-CV has opened

D. Both dryers are in service due to low IA pressure

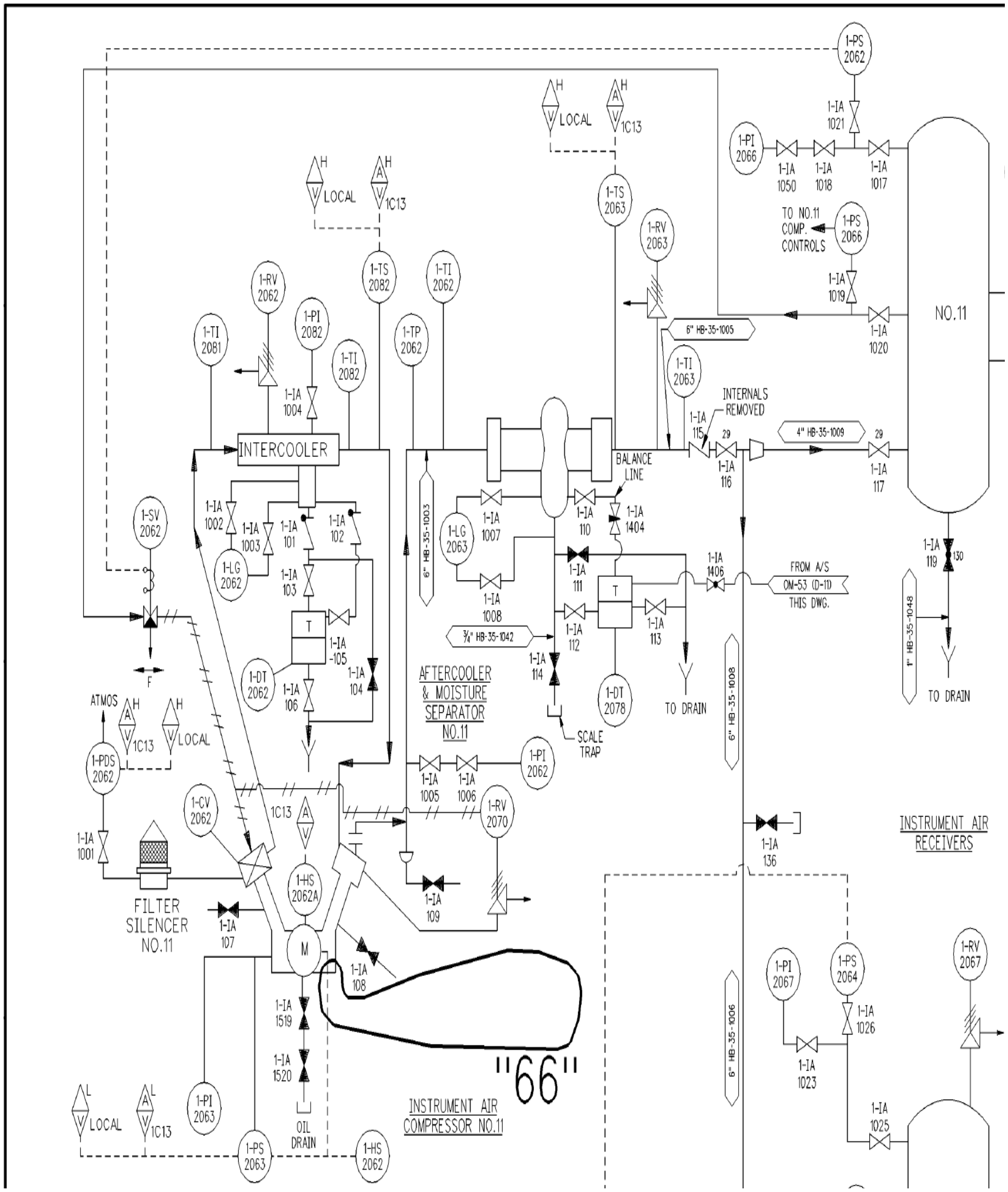
Original answer accepted was A

**Answer Explanation**
A. Correct per Alarm Manual for Window K-26
B. Incorrect, STBY compressor starts @ 93 PSIG per AOP7D section III.C notes
C. Incorrect, This CV opens @ 88 psig per AOP 7B section III C. notes
D. Incorrect, Pressure is 96 not 93 psig

**Original References**
Alarm Manual 1C13 pages 48-50
Compressed Air Lesson Plan – LOI-019-1-2 slides 62, 63

**Licensee's Justification for Change**
There is no correct answer. A candidate cannot determine if an instrument air system event has occurred solely upon 96 psig system pressure. This can be an expected condition since the instrument air compressors can cycle between 95 psig and 106 psig. Per the set point file (Attachment 1) for the Compressed Air System, pressure switches 1-PS-2062 and 1-PS-2064 cycle 11 and 12 Instrument Air Compressors between 97 and 104 psig +/- 1.86 psig to regulate instrument air system pressure.

The drawing below shows that 1-PS-2062 monitors 11 Instrument Air Receiver pressure and cycles 11 Instrument Air Compressor between the set points.




The alarm referenced in the answer explanation (INSTR AIR SYS MALFUNCTION) does not support the answer if instrument air pressure is the only parameter provided to the candidates. Indications that a loss of power to an Instrument Air Dryer cannot be based only on 96 psig instrument air system pressure. The question should have included additional data to support the correct answer.

| DEVICE | SETPOINT | WINDOW | K-26 |
|---|---|---|---|
| 1-PS-2079 | 90 PSIG (89.2 to 90.8 PSIG) | **INSTR AIR SYS MALFUNCTION** | |
| 1CR/11 Dryer malfunction | | | |
| 2CR/12 Dryer malfunction | | | |

**POSSIBLE CAUSES**

- Air Compressor failure
- Instrument Air Header break
- Instrument Air Dryers:
  - High humidity detected in bed
  - Valve switching failure (Repress, Inlet, Exhaust)
  - AMLOC probe/signal failure
  - Loss of Power to an IA dryer 5300
  - IA Pressure lowers to 93 ± 1 PSIG. 5300

**AUTOMATIC ACTIONS**

At 93 PSIG IA pressure the inservice IA Dryer de-energizes placing both dryers in service and isolating purge exhaust. 5300

**Attachment 1**

NEQR440 Setpoints By System and Instrument Number Page 3
Setpoint File For System ***019*** Unit 1 01/11

| Eqp Number / Setpt No. | SP Alias / Setpoint | Units | Direction | Action / Setpoint Tolerance | | Descr / Reset | Reset Tolerance | | Panel | Wi |
|---|---|---|---|---|---|---|---|---|---|---|
| ***1PS2008*** | | | | OPENS CONTACT | | | | | 1PNL1C13 | K- |
| 1 Des | 93 | PSIG | DEC | +1 | -1 | | | | | See R |
| Eqv | | | | | | | | | | |
| Offset N/A | Remarks | 1) Initiates 12 IA Dryer auto purge isolation | | | | | | | | |
| ***1PS2025*** | | | | OPEN CONTACTS | | | | | | |
| 1 Des | 75 | PSIG | DEC | +4 | -4 | | | | | |
| Eqv | | - - | | | | | | | | |
| Offset N/A | Remarks | Isolates Plant Air to Containment, Places Emergency Air Bottles on Service | | | | | | | | |
| ***1PS2057*** | 1XIC2054 | | | CLOSE CONTACTS | | SHUTDOWN ALARM | | | 1PNL1C13 | K- |
| 1 Des | 6 | PSIG | DEC | +1 | -1 | | | | | |
| Eqv | | | | | | | | | | |
| Offset N/A | Remarks | 1) PAC S/D. Local 1XIC2054 alarm, causes common CR alarm. Factory Set | | | | | | | | |
| ***1PS2059*** | | | | | | | | | | |
| 1 Des | 85 | PSIG | DEC | +.85 | -.85 | | | | | |
| Eqv | | - - | | | | | | | | |
| Offset N/A | Remarks | | | | | | | | | |
| ***1PS2060*** | | | | | | | | | 1PNL1C13 | K- |
| 1 Des | 90 | PSIG | DEC | +.8 | -.8 | | | | | |
| Eqv | | - - | | | | | | | | |
| Offset N/A | Remarks | | | | | | | | | |
| ***1PS2061*** | | | | | | | | | | |
| 1 Des | 88 | PSIG | DEC | +.85 | -.85 | | | | | |
| Eqv | | - - | | | | | | | | |
| Offset N/A | Remarks | | | | | | | | | |
| ***1PS2062*** | | | | | | | | | | |
| 1 Des | 97 | PSIG | DEC | +1.86 | -1.86 | 104 | +1.86 | -1.86 | | |
| Eqv | | - - | | | | | | | | |
| Offset N/A | Remarks | | | | | | | | | |
| ***1PS2063*** | | | | | | | | | 1PNL1C13 | K- |
| 1 Des | 16.21 | PSIG | DEC | +.3 | -.3 | | | | | |
| Eqv | | - - | | | | | | | | |
| Offset -1.21 | Remarks | | | | | | | | | |

**Re-grade Request**
Question #54 should be deleted due to no correct answer. It is normal for instrument air system pressure to cycle between 95 psig and 106 psig and an Instrument Air Dryer failure cannot be properly diagnosed when the only indication given is that system pressure is 96 psig.

**Question Statistics**
Question 54 was missed by 4 of 11 students. Seven candidates selected A, two selected B, and two selected D.

**Post Examination Review References**
Set point File for System 019 – Compressed Air System
1C13 Alarm Manual – Window K-26
Compressed Air System – 60712SH0001

**Licensee Addendum to Original Comments**

The question does not state that the INSTR AIR SYS MALFUNCTION alarm has annunciated or that the Air Dryer malfunction light on 1C13 is brightly lit. Without either of these indications listed, there is no correct answer. The Instrument Air Compressors normally cycle in the range of 97(+/- 1.86 psig) to 104 (+/- 1.86 psig).

Reference Alarm Manual pages 48 through 50 (attached)
Per the alarm response manual, any of three things will cause this alarm to annunciate:

1. 1-PS 2079 senses low instrument air pressure (setpoint 89.2 to 90.8 PSIG)
2. 1CR11/11 Dryer Malfunction
3. 2CR/12 Dryer Malfunction

Under AUTOMATIC ACTIONS, the description of what happens at 93 PSIG is also what happens on a loss of power to the dryer. See RESPONSE in the alarm manual for K-26, page 50. Note that the action for loss of power or lower air pressure is the same--when the cause has been corrected, reset the IA dryer.

**Air pressure below 93 PSIG or loss of power have the same effect—both towers are placed in service and regeneration purge is secured. If there is no other problem, instrument air pressure will return to normal.** See attached portions of the Compressed Air System description for a full explanation of how the Air Dryers operate in all modes**.**

| DEVICE | SETPOINT | WINDOW | K-26 |
|---|---|---|---|
| 1-PS-2079 | 90 PSIG (89.2 to 90.8 PSIG) | INSTR AIR SYS MALFUNCTION | |
| 1CR/11 Dryer malfunction | | | |
| 2CR/12 Dryer malfunction | | | |

**POSSIBLE CAUSES**

- Air Compressor failure
- Instrument Air Header break
- Instrument Air Dryers:
  - High humidity detected in bed
  - Valve switching failure (Repress, Inlet, Exhaust)
  - AMLOC probe/signal failure
  - Loss of Power to an IA dryer 5300
  - IA Pressure lowers to 93 ± 1 PSIG. 5300

**AUTOMATIC ACTIONS**

At 93 PSIG IA pressure the inservice IA Dryer de-energizes placing both dryers in service and isolating purge exhaust. 5300

(continued)

(continued) **WINDOW** **K-26**

**NOTE**

- Standby Air Compressor picks up at 93 PSIG.
- Plant Air to I/A X-CONN, 1-IA-2061-CV, opens at 88 PSIG.
- At 85 PSIG in the P/A Receiver, the P/A HDR ISOL, 1-PA-2059-CV, shuts.
- I/A back up supply 1-PCV-6301, opens at 85 PSIG.
- I/A TO CNTMT, 1-IA-2085-CV, shuts at 75 PSIG.

| CONDITION | RESPONSE |
|---|---|
| 1. Malfunction of Instrument Air System.<br><br>(continued) | 1. **PERFORM** the following:<br><br>a. **IF** the Instrument Air Header pressure is low, **THEN IMPLEMENT** AOP-7D, Loss of Instrument Air.<br><br>(continued) |

(continued) **WINDOW** **K-26**

| CONDITION | RESPONSE |
|---|---|
| 1. (continued) | 1. (continued)<br><br>b. **IF** an Air Dryer malfunction (Dryer indicating white light is brightly lit), **THEN PERFORM** the following: 5300<br><br>(1) **IF** the Malfunction was due to a loss of power **THEN PERFORM** the following:<br><br>(a) **WHEN** power has been restored **THEN RESET** the IA Dryer **PER** OI-19 INSTRUMENT AIR 5300<br><br>(2) **IF** the Malfunction was due to a loss of IA Pressure **THEN PERFORM** the following:<br><br>(a) **WHEN** IA Pressure has been restored **THEN RESET** the IA Dryer **PER** OI-19 INSTRUMENT AIR 5300<br><br>(3) **IF** the Air Dryer malfunction was **NOT** due to a loss of power **OR** loss of IA Pressure, **THEN PERFORM** the following:<br><br>(a) **BYPASS** the Air Dryer. 5300<br><br>(b) **SHIFT** to the off service Air Dryer. |

**ANNUNCIATOR COMPENSATORY ACTIONS**

**MONITOR** Instrument Air system air dryers at least once every 4 hours for normal operation.

**REFERENCES**

61082(1E-82), Sheet 1, 2, 5; 61087(1E-87), Sheet 13B; 60712(OM-53), Sheet 1

**Regrade Request**

Question #54 should be deleted due to no correct answer. It is normal for instrument air system pressure to cycle between 95 psig and 106 psig. The question did not state that any alarms were present or that the dryer malfunction light is brightly lit.. Instrument Air Dryer failure cannot be properly diagnosed when the only indication given is that system pressure is 96 psig and no alarm exists.

**NRC Response**: The licensee contends that there is no correct answer to this question because there is insufficient information provided in the stem to definitively support any of the choices with instrument air pressure decreasing to 96 psig. The NRC acknowledges that with the instrument air system at 96 psig it is still operating within its normal band. Choices B, C and D are clearly incorrect because air pressure is above the setpoints where these automatic actions would occur. To answer the question, it is not sufficient to determine that Choices B, C, and D are clearly wrong and therefore Choice A is the correct response by default. The NRC reviewed information provided by the licensee in their original post-examination comment package but was unable to confirm the effect that a loss of power would have upon the instrument air dryers (Choice A). In the additional information provided by the licensee above (and in Compressed Air System Description No. 019), the NRC now agrees that a loss of power to the dryers causes them to isolate so that there is no air bleeding out through their purge valves. Therefore, a loss of power to the dryers would not cause the instrument air system pressure to be decreasing.

**Conclusion**: The NRC agrees with the licensee's recommendation to delete this question on the grounds that there is no correct answer. The question will be deleted from the examination.

**Reactor/Senior Operator Question 60**

Each Containment Iodine Removal Unit (IRU) is ___________ capacity with each unit being ___________ efficient for removing Iodine. As humidity level approaches 99%, filter efficiency is ________________.

A. 50%, 90%, ~ 50%
B. 100%, 99%, ~ 90%
C. 100%, 90%, ~ 50%
D. 50%, 99%, ~ 90%

Original answer accepted was D

**Original Answer Explanation:**
D. Correct - (1) Each IRU is 50% capacity, with each unit being 99% efficient for removing Iodine, (2) as humidity level approaches 99%, filter efficiency is ~ 90% - Correct per EOP-5 basis

A. Incorrect, wrong efficiency @ 99% humidity

B. Incorrect, Wrong capacity

C. Incorrect, Wrong capacity and wrong efficiency

**Licensee's Justification for Change**
Answer options A and D are correct responses based on design filter capacity and efficiency. Per EOP-5 bases, efficiency ranges from 90 to 99%. Technical Specification 3.6.8 bases states that the IRUs are 50% capacity. The third part of the question is trivial information, found only in the EOP-5 basis document. There are no learning objectives or training materials that support this part of the question. No operator action is based on containment humidity changes and there is no direction to the operators in the EOP other than to start the IRUs. It is sufficient to know the capacity and efficiency of the Iodine Removal Units with the knowledge that under any conditions, two IRUs are sufficient for iodine removal.

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.8 Iodine Removal System (IRS)

BASES

**BACKGROUND**

The IRS is provided per Reference 1, Appendix 1C, Criteria 62, 63, and 64, to reduce the concentration of fission products released to the containment atmosphere following a postulated accident. The IRS would function together with the Containment Spray and Cooling Systems following a DBA to reduce the potential release of radioactive material, principally iodine, from the Containment Structure to the environment.

The IRS consists of three 50% capacity separate, independent (except for power), and redundant trains. Each train includes a moisture separator, a high efficiency particulate air filter, an activated charcoal adsorber section for removal of radioiodines, a fan, and instrumentation. The moisture separators function to reduce the moisture content of the air stream. The system initiates filtered recirculation of the containment atmosphere following receipt of a SIAS. The system design is described in Reference 1, Section 6.7.

Step: IV.W. IF THE LEAK IS INSIDE CONTAINMENT, THEN RESTORE THE CONTAINMENT ENVIRONMENT.

This block step provides steps in restoring containment environment to allow resetting of ESFAS actuation equipment. This step is consistent throughout the procedures whenever containment environment is challenged.

The EPG contains a step to provide containment radiation levels to the TSC, to evaluate the impact of potential environmental release and if containment radiation levels are high, to consider operating the Iodine Removal System. The EOP does not contain a step to provide containment radiation levels to the TSC, and contains a step to verify the Iodine Filter fans are running when addressing restoration of containment environment without requiring high containment radiation levels.

Containment radiation level assessment is performed via the Emergency Response Plan.

1. The containment iodine removal system is designed to collect within the containment the iodine released following a loss of coolant accident. Following a loss of coolant accident, a SIAS automatically starts three 50 percent capacity recirculation filter units, each with 20,000 cfm capacity. These units consist of activated charcoal filters preceded by high efficiency particulate air filters. A moisture separator is provided upstream of the particulate air filters to remove water droplets. An electric driven induced draft fan located at the end of the banks of filters pulls the containment atmosphere through these components and discharges vertically back into the containment. The operators should verify these fans have started and are operating to reduce containment iodine levels. The three containment charcoal filter units contain a total of 7300 lbs. of Barnebey-Cheney #727 coconut shell charcoal impregnated with 5 WT. % iodine compounds. Test conducted by the ORNL on Barnebey-Cheney #727 charcoal demonstrate that the installed charcoal absorbers will perform satisfactorily in removing both elemental and organic iodides for design conditions of flow, temperature and relative humidity. In these tests iodine removal efficiencies ranging between 90 to 99 percent were obtained. Filter efficiency fell toward the lower level as relative humidity approached 99 percent.

**NRC Response**: The licensee contends that either Choices A or D should be accepted as correct answers because there is no learning objective requiring the applicants to know the third item in the question (filter efficiency as containment humidity approaches 99%). The licensee provided information that supported the technical accuracy of the question and supported the one intended correct answer.

The NRC reviewed the question and determined that it matches the targeted KA as well as several other K/As. The targeted K/A was "Knowledge of the operational implications of the purpose of the charcoal filters as they apply to the containment iodine removal system (importance rating 3.1/3.4 for RO/SRO)". In addition, applicants are required to master generic K/A: 2.4.18 "Knowledge of the specific bases for EOPs (importance rating 3.3/4.0 for RO/SRO)" and system K/A 103 Containment System Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: A1.01 Containment pressure, temperature, and humidity (importance to safety 3.7/4.1 for RO/SRO).

Examination Standard ES-401 D.2.g states that "the absence of learning objective does not invalidate the question, provided that it has an appropriate K/A and technical reference." Operators should possess a level of knowledge that will allow them to assess the adequacy of various systems to perform their function and realize the operational significance in the event those systems become degraded (i.e., loss of power to one or more of the IRUs).

Prior to the exam being approved, the NRC challenged the licensee if they expected their applicants to know this information from memory. The facility representative stated that this was an expected requirement. The exam team concluded that the efficiency of the charcoal filters under design bases accident conditions is fundamentally important. If the charcoal filters were to degrade in efficiency to 50%, the containment design leakage would cause release rates to be substantially higher for a large break LOCA event and could challenge 10CFR200 limits.

Finally, the NRC will not accept an alternative answer if it is not technically correct. Answer D is the only technically correct answer.

**Conclusion**: The NRC does not agree with the licensee's recommendation to accept either Choice A or D. Choice D will remain the one correct answer because it is supported by plant reference material and the question is supported by valid K/As.

**Reactor/Senior Operator Question 61**

Unit-1 is in EOP-1 with feedwater controls in automatic mode (feedwater regulating bypass valves are controlling level) when RCP feeder breaker, 252-1201, trips. Assume no other operator action is taken. Which of the following secondary plant parameters observed ~ 25 minutes after the RCP trips indicate a loss of RCS flow is occurring?

A. Lowering steam flow and feed flow with rising S/G pressures

B. Rising steam flow and feed flow with lowering S/G pressures

C. Rising steam flow and feed flow with rising S/G pressures

D. Lowering steam flow and feed flow with lowering S/G pressures

Original answer accepted was C

**Original Answer Explanation:**
Rising steam flow and feed flow with rising S/G pressures--is correct, Tave will increase, causing ADVs to open. This will cause steam flow and feed flow to rise. S/G pressures will rise as Thot increases. Distracters are possible combinations of secondary plant parameters.

**Licensee's Justification for Change**

The stem of the question is confusing. Some candidates thought the question was asking for the indications of loss of forced flow, others thought the question asked for the indications of loss of natural circulation flow (based on "a loss of RCS flow is occurring" in the stem).

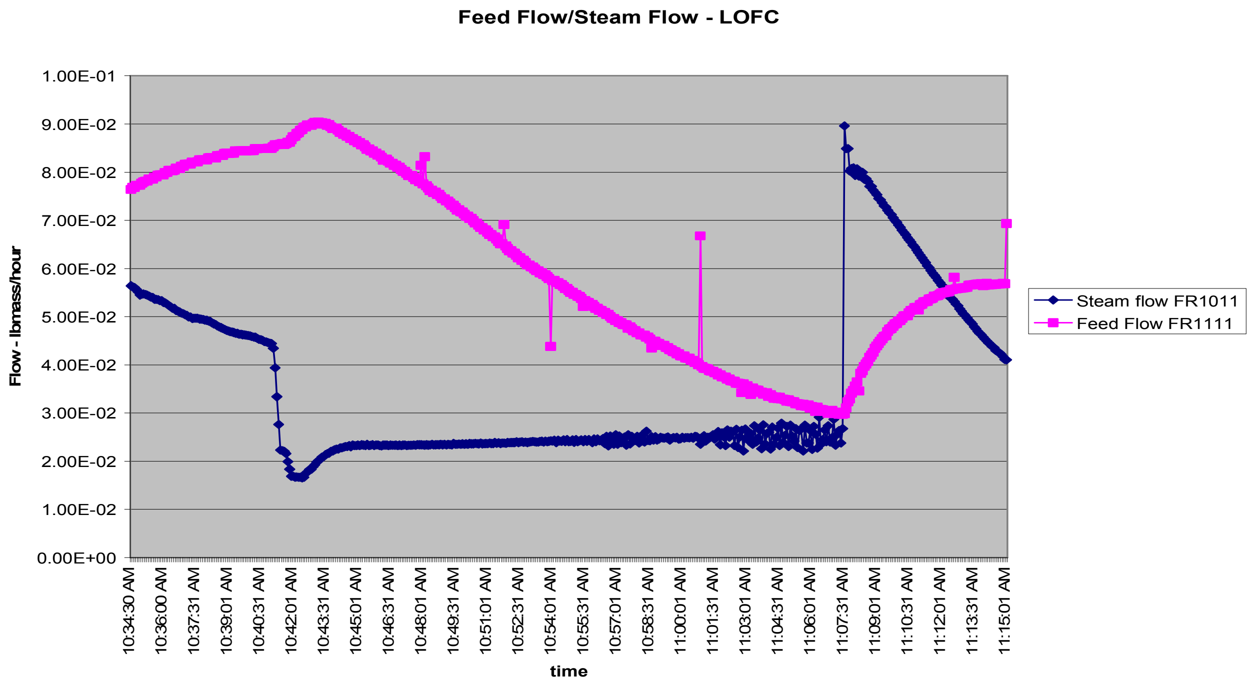
- Indications of a loss of forced flow after about 25 minutes are, the ADVs open, S/G pressure starts to lower as feed and steam flow rise (response B) until after the ADVs shut again.
- A loss of natural circulation flow would be indicated by S/G pressure rising while feed and steam flow lowered (response A, see simulator trends prior to ADVs opening. RCS flow would eventually stagnate, leaving a hot S/G at a higher pressure with feed and steam flow matched at a lower rate).

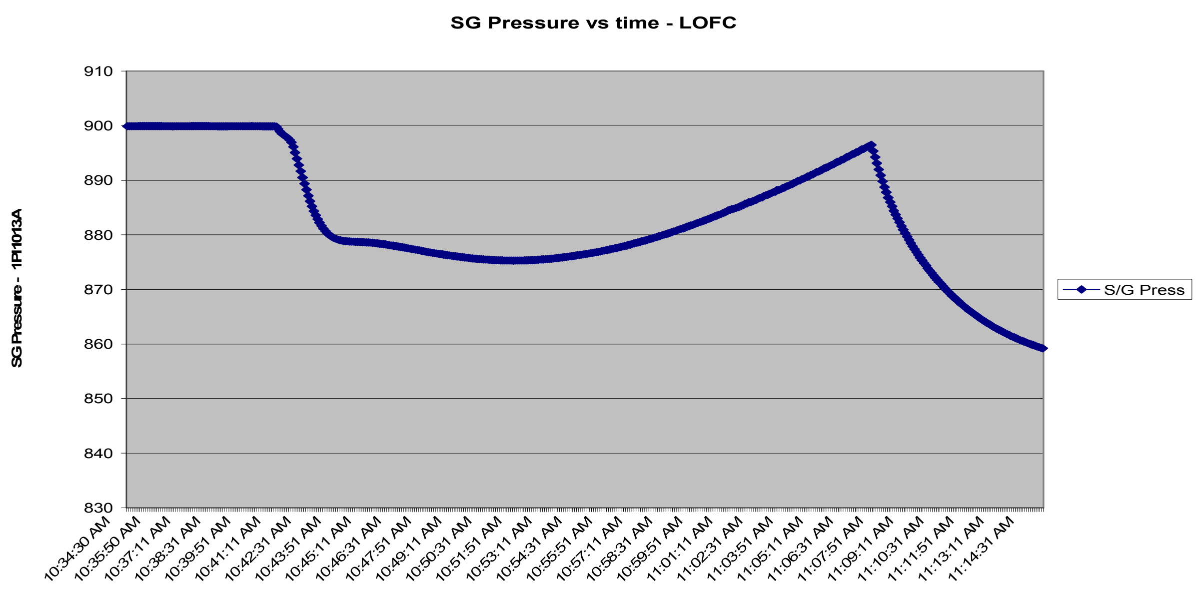
.
The original answer accepted, C, is incorrect. Steam flow and feed flow lower while S/G pressure is rising until the ADVs open. Once the ADVs open, S/G pressure lowers and feed and steam flows rise until the ADVs reshut.

D is incorrect for the time period stated. From the simulator data, feed flow would not be lowering until S/G level was greater that 0", greater than 1 hour after loss of forced flow.

No references were supplied to support the original answer explanation. See the simulator data. Initially, steam flow is dropping as decay heat is being reduced after the trip. Feed flow is rising during this time to return S/G level to 0".

**Simulator data traces:**
Feedwater is in AUTO, RCP feeder breaker trips at approximately 10:40:01 causing a loss of forced flow. Steam flow initially drops rapidly due to less heat input from the RCPs. Feed flow spikes due to shrink and then begins to drop to match steam flow. ADVs open at approximately 11:07:31 (27 1/2 minutes), steam flow spikes when ADVs open, and then starts to lower. Feed flow rises during this time. S/G pressure rises until the ADVs open. ‘A’ is also the most correct answer prior to the ADVs opening, which occurs “about 25 minutes” after loss of forced flow. The candidates were forced to make an assumption about the position of the Atmospheric Dump Valves. Answers A and B should be accepted.



**Regrade Request**
Accept responses A and B, do not accept C as a correct response.

**Question Statistics**
Question 61 was missed by 10 of 11 students. Five candidates selected A, two selected B, one selected C, and three selected D.

**Post Examination Review References**
Simulator plots of feed flow, steam flow and S/G pressure. There were no references cited for the original answer selection.

**NRC Response**: The licensee contends that there are two acceptable answers (Choices A or B) depending upon one of two assumptions: 1) there was loss of natural circulation core cooling occurring or 2) there was a loss of forced flow and the atmospheric dump valves (ADVs) were opening.

According to the facility licensee, some applicants assumed that the question was asking for indications of a loss of natural circulation because of the words in the stem "a loss of RCS flow is occurring" and thus they selected Choice A (steam and feed flow lowering with S/G pressure rising). The original intent of the question did not assume a loss of natural circulation was occurring, nevertheless, there was a choice available that matched this assumption. Choice A is an indication of gas binding in the S/G tubes which would cause a loss of natural circulation cooling and thus "indicate a loss of RCS flow is occurring."

The licensee then benchmarked this transient in the simulator to verify plant response for a loss of forced flow. The licensee provided simulator data showing secondary parameter trends over the time period indicated in the question (~ 25 minutes). The ADVs open at 27.5 minutes into this transient for less than one minute. While 27.5 minutes is close to "~25 minutes" as stated in the stem of the question, the response represents a very small time slice in the overall simulator response. The brief moment when the ADVs open is the only time that the indications contained in Choice B exist (steam and feed flow rising with S/G pressure lowering). Before and after the ADVs shut, there is no choice that correctly reflects the secondary parameter trends. However, it should be noted that the opening of an ADV is not always an indicator that there was a loss of forced flow. For example, the conditions listed in Choice B could appear following a loss of the condenser subsequent to a reactor trip as the ADV open to remove decay heat. Although not an absolute indicator of a loss of forced flow, the parameters in Choice B can exist (at the time frame specified in the question) for a loss of forced flow.

During the pre-exam review of this question, the examiners discussed this question with the examination author in the context that the trip of the RCP feeder breaker 252-1201 caused only a single RCP to trip. At no time during question review discussions was there an indication that all four RCPs had tripped. Using this assumption, Choice C could be a possible correct answer based upon secondary parameter response with the loss of only one RCP.

However, in the post-exam comments, the licensee indicated that tripping this feeder breaker interrupted power to all four RCPs. Given the information that all four RCPs had tripped, the question now contains an editorial error that could introduce confusion. As stated above, tripping the “RCP feeder breaker, 252-1201,” causes all RCPs to trip, not a single RCP. However, the final stem statement is: “Which of the following secondary plant parameters observed ~ 25 minutes after **the RCP trips** indicate a loss of RCS flow is occurring?” This statement indicates that a single RCP has tripped. If the applicants had not memorized the breaker designation for the RCP feeder breaker, they could conclude that three RCPs were still running. If the applicants think that three RCPs are running, then Choice C could be a correct answer depending on the assumptions made regarding the decline of decay heat, the amount of pump heat added and the operation of the turbine bypass valves.

Applicants could make various assumptions that caused them to choose either A, B or even C as the answer. In addition, it is not expected that applicants would have memorized the function of RCP feeder breaker 252-1201 based solely on the breaker designation. The stem should have either supplied the information regarding the function of the RCP feeder breaker or should not have stated that only one RCP tripped. (The source of this question was listed as the CCNP exam bank. A similar question had been used on the 2002 Calvert Cliffs NRC exam however, the stem had been altered. The question on the 2002 NRC exam had listed “the RCP feeder breaker” as the “RCP **bus** feeder breaker” (emphasis added), thereby clarifying the function of this breaker.)

Also, it should be noted that all of the parameter trends in Choices A and B are opposite of each other. Examination Standard ES-403 D.1.c states that in cases where “both answers contain conflicting information, the question will likely be deleted.” In addition, the additional problems with stem clarity cause this question to be fundamentally flawed.

**Conclusion**: The NRC does not agree with the licensee’s recommendation to accept either A or B. Given the ambiguity of the question regarding what is meant by “loss of RCS flow,” the variation of secondary parameters over the “~ 25 minutes” of this transient, the disparity in the secondary parameter trends in the two recommended answers (A and B), and the editorial error regarding the number of RCPs which had tripped, this question will be deleted.

**Reactor/Senior Operator Question 68**

Unit 2 has just completed a refueling outage and is conducting PSTP3," Escalation to Power Test Procedure", to test at the power plateau of 85% power. At 80% it was determined that Frt is greater than the full power value of T.S 3.2.3. While reviewing the data a transient occurs and power rises to 90% and is stabilized. Which of the following is required?

A. Reduce Thermal Power to less than or equal to 85% within 1hour

B. Reduce Thermal Power to less than or equal to 85% within 15 minutes

C. Reduce Thermal Power to less than or equal to 80% within 1 hour

D. Reduce Thermal Power to less than or equal to 80% within 15 minutes

Original answer accepted was B

**Original Answer Explanation**

A. Incorrect, the power level is correct but the time to reduce is wrong
B. Reduce Thermal Power to less than or equal to 85% within 15 Mins - Correct per T.S. 3.1.8
C. Incorrect, the time to reduce power and the power level are wrong
D. Incorrect, the power level to reduce to is incorrect

**Original References**

Technical Specification 3.1.8, Special Test Exception (STE) – Modes 1 and 2
PSTP-3, Escalation to Power Test Procedure

**Licensee's Justification for Change**

Candidates were not provided with enough information to properly test their ability to implement the special test exception of Technical Specification 3.1.8. The question states at 80% it was determined that $F_r^T$ is greater than the full power value of T.S. 3.2.3 and a transient occurred which raised power to 90% and stabilized.

The question did not state that Special Test Exception 3.1.8 was invoked. Since no Technical Specifications were being violated, there is no reason to assume the Special Test Exception, 3.1.8 applied. At 80% power, $F_r^T$ can exceed the 100% power limit and not exceed the limit for 80 or 90% (See figure 3.2.3). The intent of the question was to test candidate knowledge of test exception of T.S. 3.1.8. The PSTP-3 procedure does not specifically direct invoking the special test exception of T.S. 3.1.8. Where 3.1.8 is invoked, procedures have a step to ensure the requirements are met.
See PSTP-2 attachment 3 as an example. No similar document exists in PSTP-3. The stem of the question should have stated that the special test exception had been invoked. Since the special test exception was not applicable, maintaining power stable at 90% would be the correct response. There would be no reason to subject the plant to an additional transient.

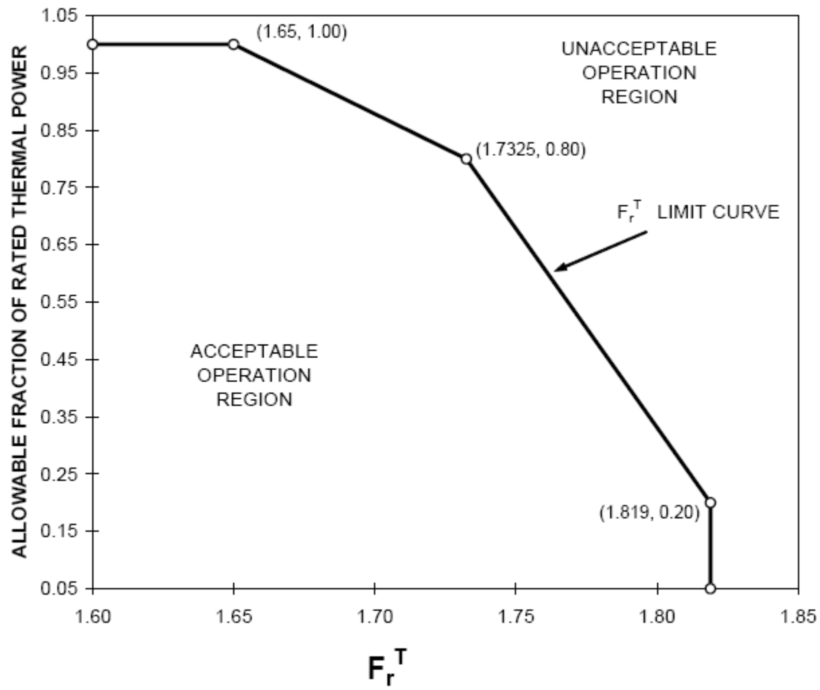



**Figure 3.2.3**

**Total Integrated Radial Peaking Factor ($F_r^T$) vs.**
**Allowable Fraction of Rated Thermal Power**

**While operating with $F_r^T$ greater than 1.65, withdraw CEAs to or above the Long Term Steady State Insertion Limits (Figure 3.1.6)**

**ATTACHMENT 3**
**SPECIAL TEST EXCEPTION VERIFICATION**

Seq #________

Technical Specification SR 3.1.8.1 requires that power level be checked once per hour to verify that the highest reading WRLC is less than 1% power. An independent reviewer is required to initial in the "VER." column.

| DATE | TIME | POWER LEVEL IN RANGE? | INIT. | VER. |
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| DATE | TIME | POWER LEVEL IN RANGE? | INIT. | VER. |
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3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Special Test Exception (STE)-MODES 1 and 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC);"
LCO 3.1.4, "Control Element Assembly (CEA) Alignment;"
LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits;"
LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits;"
LCO 3.2.2, "Total Planar Radial Peaking Factor ($F^T_{xy}$);"
LCO 3.2.3, "Total Integrated Radial Peaking Factor ($F^T_r$);" and
LCO 3.2.4, "AZIMUTHAL POWER TILT ($T_q$)"

may be suspended, provided THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Test power plateau exceeded. | A.1 Reduce THERMAL POWER to less than or equal to test power plateau. | 15 minutes |

The original reference used to develop the question was PSTP-3, Escalation to Power Test Procedure applicability/scope statement 2.6. This is a boiler-plate statement in all the PSTPs. In practice, 3.1.8 is not entered during the performance of PSTP-3. Step 6.7 also has a "90% limiting" parenthetical clause which indicates that, at least for this section of the procedure, 3.1.8 cannot apply.
Applicability/scope statement 2.6 states PSTP-3 shall be considered as physics testing but does not direct invoking the special test exception of Technical Specification 3.1.8. The statement only mentions that the special test exception may apply under specific conditions.

ESCALATION TO POWER TEST PROCEDURE — PSTP-3 Rev. 30 Page 6 of 93

2.6 This procedure shall be considered PHYSICS TESTS and certain SPECIAL TEST EXCEPTIONS of Technical Specification 3.1.8 may apply as follows:

- If measured MODERATOR TEMPERATURE COEFFICIENT (MTC) and/or the Peaking Factors $F_r^T$ and $F_{xy}^T$ do not meet the limits of Technical Specifications 3.1.3, 3.2.2 and 3.2.3 early in cycle life, SPECIAL TEST EXCEPTION 3.1.8 may be invoked below 85% Rated Thermal Power (RTP).
- Thermal Power shall be restricted to less than 85% Rated Thermal Power per Technical Specification Surveillance Applicability 3.1.8. **[B-14]**
- Technical Specification Surveillance Requirement 3.1.8.1 is satisfied by recording Thermal Power at least once per hour, ensuring that power remains less than or equal to the test power plateau.

In accordance with PSTP-3 $F_r^T$ is evaluated for technical specification compliance at step 6.7, Power increase to 85% RTP. PSTP -3 section 6.7 does not direct invoking the special test exception, nor is it referred to at any step in the procedure.

**ESCALATION TO POWER TEST PROCEDURE** — **PSTP-3 Rev. 30 Page 25 of 93**

**6.7 Power increase to 85% RTP (85% RTP NOMINAL, 90% RTP LIMITING)**

A. **COMMENCE** power increase to 85% RTP observing limits of PRECAUTION 5.2 **AND CONTINUE** to monitor the power increase by recording values on Attachment PSTP-3-5 hourly and Attachment PSTP-3-4 **PER** Step 6.2.B.

B. **PRIOR** to exceeding 70% RTP, **VERIFY** that the corrected predicted HFP MTC is less than the most positive value allowed at 100% RTP per Technical Specification 3.1.3 and T.S. Figure 3.1.3-1 **OR ESTABLISH** a maximum boron concentration to ensure compliance **AND RECORD** reference below. **MARK** the MAXIUMUM BORON CONCENTRATION "N/A", **IF** such a restriction is not required.

MAXIMUM BORON CONCENTRATION (if required): _______ ppm

Reference:_______________________________________________

___________ SE

C. **PRIOR** to exceeding 70% RTP, **VERIFY** $F_r^T$, $F_{xy}^T$, and $T_q$ are within their Technical Specification limits. **[B-25]**

___________ SE

At 85% reactor power $F_r^T$ is evaluated again for Technical Specification 3.2.3 compliance per Appendix B.

**6.8 After reaching steady-state operating conditions at 85% RTP (85% RTP NOMINAL, 90% RTP LIMITING)**

A. **DIRECT** Operations to perform OI-30 calibration.

__________
SE

B. **PERFORM** a power distribution measurement in accordance with Appendix B, step 6.3.

__________
SE

C. **VERIFY** per the Operator's Log Sheets that **DELTA T POT SETTINGS** are satisfactory for operation above 90% RTP.

__________
SE

**6.8 85% RTP (Continued)**

D. **PRIOR** TO EXCEEDING 85% RTP, **ENSURE** all Review Criteria have been met **OR** reviewed by PORC and approved by the Plant General Manager. **[B-23]**

PORC Meeting Number (if applicable): ________________________

__________
SE

PSTP-3 Appendix B 6.3 log measured $F_r^T$ value and compares its value against the technical specification limit. If the current value of $F_r^T$ is larger than the full power limit the procedure directs notifying the Principal Engineer- Fuel Services Unit. PSTP-3 does not direct invoking Technical Specification 3.1.8 Special Test Exception if the $F_r^T$ limits are exceeded.

**Appendix B, CECOR Library Qualification and Power Distribution Measurement**
**Page 8 of 12**

**6.3 85% Power Plateau**

A. **BLOCK** the periodic CECOR execution by performing the following steps from the System level on the plant computer:

1. From the Main Menu, **SELECT** "System Tasks."
2. **SELECT** "Point Editor."
3. **SELECT** "Edit a Point."
4. **ENTER** Point ID CEPERIOD.
5. **SET** the value of CEPERIOD to 1 (one).

_____________ SE

> **NOTE**
> Step 6.3.B may be repeated as necessary.

B. **OBTAIN** a corefollow CECOR, option 1.

CECOR Printout Date/Time __________/__________

_____________ SE

C. **PERFORM** a power distribution comparison per APPENDIX F.

_____________ SE

> **NOTE**
> Step 6.3.D through 6.3.F may be performed concurrently.

D. **COMPARE** $F_r^T$, $F_{xy}^T$, and $T_q$ from step 6.3.B with the following acceptance criteria **AND DOCUMENT** on Attachment A-1 and below.

| | MEASURED | ACCEPTANCE CRITERIA (TS Value - current pwr level) |
|---|---|---|
| $F_{xy}^T$ | ________ | ≤ ______________ (TS 3.2.2) |
| $F_r^T$ | ________ | ≤ ______________ (TS 3.2.3) |
| $T_q$ Upper | ________ | ≤ 0.030 (TS 3.2.4) |
| $T_q$ Lower | ________ | ≤ 0.030 (TS 3.2.4) |

_____________ SE

G. **IF** the current value of either $F_r^T$ or $F_{xy}^T$ is larger than the full power limit, **THEN NOTIFY** the PE-FOSU. **[B-91]**

_____________ SE

### Regrade Request

Question # 68 be should be deleted. As written, there is no correct answer. The question stem did not specifically state the special test exception had been invoked. The candidate could not assume that the requirements of T.S. 3.1.8 applied.

### Question Statistics

Question 68 was missed by 11 of 11 candidates. One candidate selected A, 5 candidates selected C, 5 selected D.

**Post Examination Review References**
Technical Specification 3.1.8 - Special Test Exception – Modes 1 and 2
Technical Specification 3.2.3 – Total Integrated Radial Peaking Factor
PSTP-3 – Escalation to Power Test Procedure

**NRC Response**: The licensee contends that the question should be deleted because insufficient information was provided to the applicants and because the question did not specifically state that the special test exception of T.S 3.1.8 was being invoked.

The question indicated that "At 80% power it was determined that Frt is greater than the full power value of T.S. 3.2.3." When the question was developed, the author and examiners concluded that if the value for $F_r^T$ exceeded the 100% power limit that it would have also exceeded the 80% and 90% power limits. However, the licensee correctly states, based upon COLR Figure 3.2.3 above, that at 80% power, $F_r^T$ can exceed the 100% power limit and not exceed the limit for 80 or 90% because it is a flux ratio, not an absolute value. $F_r^T$ would have to exceed 1.7325 in order for tech spec 3.2.3 to apply at 80%. At 100% power, $F_r^T$ would only have to exceed 1.65 in order to have exceeded the "full power value of T.S. 3.2.3".Thus, with the plant at 90% power, T.S. 3.2.3 may or may not be exceeded depending upon the value of $F_r^T$. Since no value was provided for $F_r^T$, nor was Figure 3.2.3 provided, it was impossible to answer the question from the information given.

Regarding the contention that special test exception in Tech Spec 3.2.3 did not apply, the NRC cannot reach a conclusion that the licensee's position is substantiated. The documentation provided is ambiguous regarding this contention. However, it is not necessary to reach a determination on this issue in order to determine that there is no correct answer.

**Conclusion**: The NRC agrees with the licensee's recommendation to delete the question due to insufficient information provided to the applicants. The question will be deleted.

**Senior Operator Question 10**

A large break LOCA has occurred on Unit-2 and all RCPs have been tripped. The RO is attempting to verify subcooled natural circulation and reports the following:

Pressurizer Pressure is 150 PSIA being maintained by HPSI & LPSI flow
RCS Subcooling based on CETs is 5°F

Which one of the following set of conditions is the **minimum** needed to ensure adequate core cooling?

A. HPSI and LPSI flow appropriate for current RCS pressure AND Thot ~ 425°F

B. HPSI and LPSI flow appropriate for current RCS pressure AND Thot ~ 405°F

C. HPSI and LPSI flow appropriate for current RCS pressure AND Thot ~ 388°F

D. HPSI and LPSI flow appropriate for current RCS pressure AND Thot ~ 360°F

Original answer accepted was B

**Original Answer Explanation**

Need to recognize that with CETs at 5°F subcooling, subcooled natural circulation is not being met. Per EOP-5 Block Step IV. N 2, for verifying subcooled natural circulation , if natural circulation subcooling is not being met, then need to ensure no more than 50° superheat to ensure adequate core cooling.

Since RCS pressure is 150 PSIA the **minimum** conditions for providing at less than 50°F superheat
A. 425°F would not provide < 50°F

B. HPSI and LPSI flow appropriate for current RCS pressure AND Thot ~ 405°F - Correct would give < 50°F superheat (Sat temp for 150 PSIA = 358.4°F)

C. 388°F would provide < 50°F but the question asked the minimum conditions to give < 50° superheat

D. 360°F would provide < 50°F but the question asked the minimum conditions to give < 50° superheat

**Licensee's Justification for Change**
The stem of the question is confusing. Candidates were confused about the meaning of “Which one of the following set of conditions is the **minimum**”. EOP-5 basis states that flow out the break is the heat removal process for a large break LOCA. EOP Attachment 10 has a chart that indicates adequate HPSI/LPSI flow for heat removal after a LOCA. Candidates reasoned that core cooling was met by HPSI and LPSI flow and heat removal is adequate if subcooling exists.

The data contained in the question informs the candidate that CET subcooling is 5°F. The intent of the question was to evaluate the candidates' ability to recall the alternate actions for EOP-5 section IV.N.

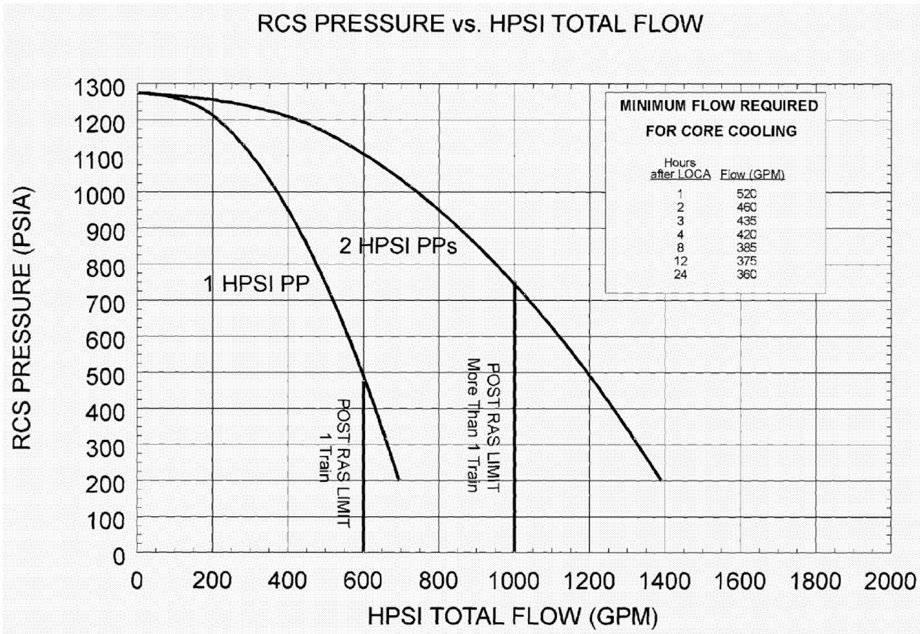
The question asks the SRO candidates to determine which set of conditions is the **minimum** needed to ensure adequate cooling. Each possible selection informs the candidates that HPSI and LPSI flow were appropriate, which satisfies one component of adequate core cooling. In addition to the HPSI and LPSI flow information, each possible selection gave various $T_{hot}$ values. Six of the seven SRO candidates interpreted the meaning of "minimum conditions" in the question stem other than what was intended.

Their understanding of "minimum conditions" guided their thought process to select the lowest superheated temperature. Six SRO candidates selected D as the correct response since $T_{hot}$ at 360°F is only 1.57°F above the saturation temperature for 150 psia. The RCS saturation temperature for 150 psia is 358.43°F. This value was interpreted as the **minimum** RCS condition needed to ensure adequate RCS core cooling. The intent of the question was for the candidates to interpret "minimum condition" as the maximum degree of superheat without exceeding the 50°F limit. One SRO candidate interpreted the question as intended and selected B as the correct response.

From EOP-5 bases-

> Small and large break LOCAs differ in their effect on the post-LOCA RCS heat removal process. For a large break, the only path necessary for RCS heat removal in both the short and long term is the break flow with core boiloff. For small breaks, heat removal via the flow out the break is not sufficient to provide cooling and, therefore, steam generator heat removal is required. The procedure takes this into account with the decisions that must be made. Although distinct small and large break LOCA information is contained in the basis section of this procedure, the action steps to be used during the actual emergency do not require the operator to distinguish between break sizes.

ATTACHMENT(10)
Page 1 of 1
HIGH PRESSURE SAFETY INJECTION FLOW



EOP-5
Rev 22/Unit 1
Page 40 of 97

**IV. ACTIONS**

| RECOVERY ACTIONS | ALTERNATE ACTIONS |
|---|---|
| N. MAINTAIN RCS FLOW VERIFICATION. | |
| 1. **IF ANY** RCPs are running, **THEN** verify $T_{HOT}$ minus $T_{COLD}$ is less than 10° F in the loop(s) with the unaffected S/G. | 1.1 **IF** $T_{HOT}$ minus $T_{COLD}$ is greater than 10° F in the loop(s) with the unaffected S/G, **THEN** trip **ALL** RCPs. |
| **NOTE**<br>Verification of RCS temperature response to a plant change during natural circulation takes approximately 5 to 15 minutes following the action due to increased loop cycle times. | |
| 2. **IF ALL** RCPs have been secured, **THEN** verify subcooled natural circulation by the following:<br>• RCS subcooling is at least 30° F based on CET temperatures<br>• $T_{HOT}$ minus $T_{COLD}$ less than 50° F<br>• $T_{COLD}$ constant or lowering<br>• $T_{HOT}$ constant or lowering<br>• CET temperatures trend consistent with $T_{HOT}$<br>• Steaming rate affects RCS temperatures | 2.1 **IF** subcooled natural circulation can **NOT** be verified, **THEN** verify adequate RCS cooling flow by the following:<br>• **ALL** available CHG PPs are operating<br>• SIS flow is appropriate **PER** ATTACHMENT(10), HIGH PRESSURE SAFETY INJECTION FLOW, **AND** ATTACHMENT(11), LOW PRESSURE SAFETY INJECTION FLOW<br>• At least ONE S/G available for heat removal<br>  • S/G level greater than (-)170 inches<br>  • capable of being supplied with feedwater<br>  • capable of being steamed<br>• CET temperatures are less than 50° F superheated |

**Regrade Request**

Due to unclear wording of the question stem, selections B and D should be accepted as correct responses.

**Question Statistics**

Question SRO 10 was missed by 6 of 7 SRO candidates. One candidate selected B, six candidates selected D.

**Justification References**

EOP-5 – Loss of Coolant Accident
EOP-5 – Loss of Coolant Accident Technical basis
EOP Attachments
Steam Tables Properties of Saturated and Superheated Steam – Table 2

**NRC Response**: The licensee contends that the word "**minimum**" introduces ambiguity into the question. Based upon an applicant's thought process "minimum" could refer to a condition that is just barely acceptable for ensuring adequate core cooling exists (Choice B) or "minimum" could refer to a condition with the lowest superheated temperature (Choice D) and is thus the safest condition. There were no questions from the applicants pertaining to this question during the examination administration. Therefore, there was no opportunity to identify this ambiguity at the time of examination administration.

The NRC challenged the use of the word "minimum" during the exam review. The licensee stated that applicants were exposed to this type of question during the training and there would be no ambiguity. The NRC accepted this assertion at face value because of the insistence of the licensee representative. In hindsight, the NRC acknowledges how the word minimum could be misinterpreted to lead the applicant to the lowest temperature (versus closest temperature) that ensured adequate core cooling.

**Conclusion**: The NRC agrees with the licensee's comment that the question's intent was ambiguous. Therefore, the NRC will accept either Choice B or D as correct answers.

**Senior Operator Question 15**

Using provided reference:
Unit 1 was operating at 100% power when a large Loss of Coolant Accident (LOCA) occurred. EOP-5 has been implemented. Hydrogen concentration rose to .5% and the Hydrogen Recombiners were started. CNTMT TEMP prior to the event was 90°F. Two hours have passed since the Hydrogen Recombiners were started and now the following conditions exist:

H2 concentrations is now .8% and rising
11 Recombiner power setting is 50 KW
12 Recombiner is OFF
Containment Pressure is 4.5 PSIG

Which of the following is the correct action?

A. Set 11 Hydrogen Recombiner power setting to 57 KW

B. Set 11 Hydrogen Recombiner power setting to 60 KW

C. Set 11 Hydrogen Recombiner power setting to 63 KW

D. Set 11 Hydrogen Recombiner power setting to 65 KW

Original answer accepted was B

**Original Answer Explanation:**
Per the graph of OI-41A with a Cntmt Press at the CSAS set point of 4.25 psig which gives a KW of 60.5 KW. Per the EOP-5 basis document within 1 hour of starting the recombiner it should be functioning, and one recombiner is designed to reduce H2 concentration faster than can be produced from a design basis accident, so if set properly then the H2 concentration should be lowering 2 hours after the recombiner was started. The fact that H2 concentration has risen should indicate that the recombiner is not functioning properly.

A. Set 11 Hydrogen Recombiner power setting to 57 KW-- Is Incorrect for the conditions given this setting is to low.

B. Set 11 Hydrogen Recombiner power setting to 60 KW - Is correct. for the conditions given.

C. Set 11 Hydrogen Recombiner power setting to 63 KW-- Is incorrect for the conditions given, this setting is too high.

D. Set 11 Hydrogen Recombiner power setting to 65 KW-- IS incorrect for the conditions given this setting is to high

**Licensee's Justification for Change**
This question required the applicants to determine the correct power setting for the recombiner based on initial containment temperature and current containment pressure. The correct answer per the Key was "B" which corresponds to 60 KW. Looking at the reference provided (Figure 10 from OI-41), to obtain the answer the applicants had to extrapolate a power setting based on a given containment pressure of 4.5 psig. The line chosen indicates a power of ~ 61 KW.

1. The H2 recombiner would be started per step G.9 of EOP-5 which directs starting per OI-41A
2. Per OI-41A, pg 6 (attached) the candidate would set the recombiner power to the level determined by adjusting the potentiometer and observing the power meter.
3. Neither answer is directly at the power setting indicated on the graph.
4. The power meter increments are 2 KW. It is difficult to read increments of 1 KW on these meters, (attached photo), an acceptable setting for these conditions could be 60-63 based on acceptable tolerances for reading the meter increments.
5. Additionally, the Hydrogen recombiner technical manual, section 4.5.1.7 (attached), states the power setting should be set to maintain a temperature of 1200° F ± 25°F. It also states that power adjustment required are approximately 4 KW per 75°F temperature change.
6. Using the data from # 5 above, the range of power settings associated with a ± 25°F tolerance from the ~61 KW setting would be approximately (4KW/75°F = .0533 X 25 = 1.33 KW). This gives an approximate range of 59.67 – 62.33 KW.
7. 61 was not a selection, it is reasonable that a candidate would choose 60 KW, or the more conservative value of 63 KW, and still be in compliance with the technical manual for the operation of the recombiner, and acceptable for operation.

From Technical Manual

4.5.1.2 Startup - Energize the power supply by closing Breaker No. (this number to be supplied by Customer).

4.5.1.3 The "PWR in Avail" light will be actuated on th control panel.

4.5.1.4 Set the "PWR Out SW" on the control panel to th "ON" position. The light on the switch will be activated.

4.5.1.5 Gradually turn the control potentiometer labele "PWR ADJ" until 48 KW is indicated on the power meter which is labeled "PWR OUT". (Note there a lag in the meter reading so turn the potentio meter knob slowly.)

4.5.1.6 Hold power at 48 KW for 5 hours and then read recombiner temperature.

4.5.1.7 If temperature is not 1200 ± 25°F adjust power to bring temperature into this range. Power adjustment required is approximately 4 KW per 75°F temperature change. Allow recombiner to stabilize for two hours after each power adjustment.

4.5.1.8 When recombiner temperature fo 1200 ± 25°F has been obtained, record recombiner power from readout of the power meter and record containment temperature and containment pressure from plant instruments.

4.5.1.9 Determine calibration factor (cc) from the Recombiner Power Correction Factor vs. Calibration Temperature Curve (see Page 22).

 JAN 2 8 1985

Attachment

## 6.0 PERFORMANCE

### 6.1 HYDROGEN RECOMBINER STARTUP (11, 12, 21, 22)

#### A. Initial Conditions

1. It is desired to place hydrogen recombiners in service for testing **OR** post accident conditions. 1002

#### B. Procedure

1. **ENSURE** the desired hydrogen recombiner heater control potentiometer is set at 000: 1002
   - 11 $H_2$ RECOMBINER, 1-HS-7501
   - 12 $H_2$ RECOMBINER, 1-HS-7506
   - 21 $H_2$ RECOMBINER, 2-HS-7501
   - 22 $H_2$ RECOMBINER, 2-HS-7506
2. **IF** starting the hydrogen recombiner due to an accident, **THEN PERFORM** the following:
   a. **OBTAIN** pre-accident Containment temperature from shift log readings.
   b. **OBTAIN** current Containment pressure reading.
   c. Using Figure 1, **DETERMINE** the required power (KW).
   d. **PLACE** the desired hydrogen recombiner(s) ON/OFF handswitch(es) in ON: 1002
      - 11 $H_2$ RECOMBINER, 1-HS-7502
      - 12 $H_2$ RECOMBINER, 1-HS-7507
      - 21 $H_2$ RECOMBINER, 2-HS-7502
      - 22 $H_2$ RECOMBINER, 2-HS-7507

6.1.B.2 **Procedure** **(Continued)**

| **CAUTION** |
|---|
| Maximum power to a hydrogen recombiner is limited to 75 KW. |

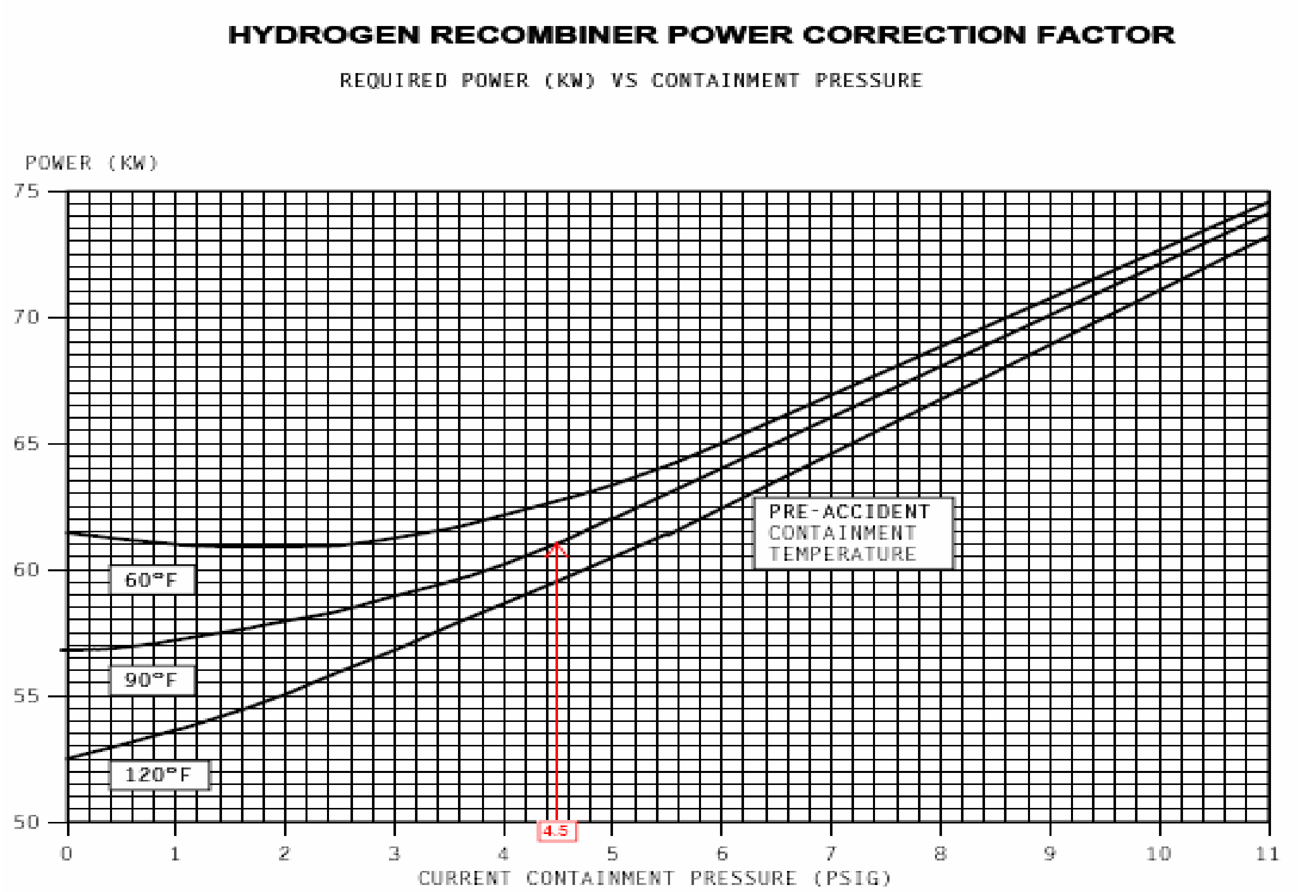
1002

e. **RAISE** the applicable hydrogen recombiner power to the level determined in Step 2.c, by adjusting its Heater Control Potentiometer: 1002

- 11 $H_2$ RECOMBINER, 1-HS-7501
- 12 $H_2$ RECOMBINER, 1-HS-7506
- 21 $H_2$ RECOMBINER, 2-HS-7501
- 22 $H_2$ RECOMBINER, 2-HS-7506

f. **MONITOR** the hydrogen recombiner(s) placed in service for proper operation by checking each applicable power meter (wattmeter) indicates the value determined in Step 2.c:

- 11 $H_2$ RECOMBINER, 1-XI-7501
- 12 $H_2$ RECOMBINER, 1-XI-7506
- 21 $H_2$ RECOMBINER, 2-XI-7501
- 22 $H_2$ RECOMBINER, 2-XI-7506


HYDROGEN RECOMBINER POWER CORRECTION FACTOR
REQUIRED POWER (KW) VS CONTAINMENT PRESSURE
POWER (KW)
75
70
65
60
55
50
60°F
90°F
120°F
PRE-ACCIDENT CONTAINMENT TEMPERATURE
4.5
0
1
2
3
4
5
6
7
8
9
10
11
CURRENT CONTAINMENT PRESSURE (PSIG)


11 H2 RECOMBINER
WATTS
1-XI-7501
TEMP
1-TI-7500
HTR CONTR
1-HS-7501
T/C SEL
1-HS-7500

**Regrade Request**
The range of power settings associated with a ± 25°F tolerance from the ~61 KW setting would be approximately (4KW/75°F = .0533 X 25 = 1.333 KW). This gives an approximate range of 59.67 – 62.33 KW. B and C should be accepted for question #15 on the SRO exam.

**Question Statistics**
Question SRO 15 was missed by 4 of 7 candidates. Three candidates selected B, four selected C.

**Justification References**
OI-41A
Hydrogen Recombiner technical manual

**NRC Response**: The licensee contends that because the value determined from the graph falls between Choices B or C that either one should be accepted as a correct answer. The NRC acknowledges that there was an error in the development of the question in that one of the choices should have been a hydrogen recombiner power setting of 61 KW. This value, as stated by the licensee, falls between Choices B and C which are 60 KW and 63 KW, respectively. This left the applicants in a dilemma of choosing between an answer that is closest to, but slightly less than, the value determined from the graph or an answer that is above the determined value but ensures that the minimum power is being applied to the recombiner. Despite the absence of a correct answer among the choices, there were no questions raised by the applicants.

Information provided by the licensee indicates that there is a tolerance associated with the power setting. Using 61 KW as a reference point, 60 KW is within the lower band of the expected tolerance (± 1.33 KW) whereas 63 KW is not. However, the licensee also suggests that based upon the incremental values on the power meter (2 KW), 63 KW would be within an acceptable band. But regardless of the ability of an operator to read the power setting meter, the target setting was to be 61KW.

In the final analysis, there was no correct answer provided among the choices and thus it should be deleted from the examination.

**Conclusion**: The NRC disagrees with the licensee's recommendation to accept either Choice B or C as a correct response. The question was flawed in that there was no correct answer provided among the choices. The question will be deleted from the examination.

**Senior Operator Question 24**

Unit 2 is in Mode 6 with refueling in progress and Normal Containment Purge in service. The Equipment hatch is installed and the Personnel Airlock (PAL) is open. A momentary loss of power causes the operating Main Exhaust Fan to trip.
(a) What is the effect on containment parameters, (b) What is the correct action?

A. (a) Containment refueling pool level decreases, (b) Continue refueling operations.

B. (a) Containment pressure rises 1 to 2 PSIG, (b) Initiate additional containment cooling

C. (a) Containment area radiation monitors (RE-5316-A through -D) indicate higher, (b) Start all available Iodine Filter Units

D. (a) Containment refueling pool level increases, (b) Continue refueling operations.

Original answer accepted was A

**Original Answer Explanation: (***Note that the original answer justification is incorrect. The responses do not match the exam responses, D is listed as correct in the justification and does not match the wording of response A*)

A. Containment pressure rises 1 to 2 PSIG--incorrect, containment pressure will change, but experience indicates, the change will be less than .5 PSIG.

B. Area radiation monitors (RE-5316A-D) indicate higher--incorrect, the area monitors would not change if Purge is lost.

C. Refueling pool level increases--Incorrect The Main Exhaust Fan tripping would cause Containment Purge to secure. This would cause containment pressure to rise slightly, with the transfer tube gate valve open, refueling pool level will decrease (Not Increase) accordingly due to the differential pressure between the SFP area and containment.(SFP is maintained at a slight negative pressure)

D. Refueling pool level increases--**Is Correct**. The Main Exhaust Fan tripping would cause Containment Purge to secure. This would cause containment pressure to rise slightly, with the transfer tube gate valve open, refueling pool level will decrease accordingly due to the differential pressure between the SFP area and containment. (SFP is maintained at a slight negative pressure)Continue refueling operations since no loss of RFP level.

per OI-36 general precaution F. The Main Exhaust Fan tripping would cause Containment Purge to secure which would cause a change in the differential pressure between the SFP and the RFP

**Licensee's Justification for Change**
There is no correct answer. The question asks: (a) What is the effect on containment parameters, (b) What is the correct action?
The Effect on containment parameters is there will be a small containment pressure rise due to the Purge Exhaust and Supply fans tripping. This increase in pressure will cause Refuel Pool water to shift to the Spent Fuel Pool through the open transfer tube. The net result being, Containment Pressure increases and Refuel Pool Level lowers.

The correct action would be to suspend fuel handling operations due to the loss of the Auxiliary Building and Waste Processing Supply Fan, which is required by plant procedures.

From OI-22A pg. 5, (see prints also)

B. **The following fans will trip if the only operating Main Exhaust Fan on Unit 2 is secured:**

- Unit 2 Containment Purge Supply and Exhaust Fan
- 21 Auxiliary Building and Waste Processing Supply Fan
- 21 AND 22 Auxiliary Building and Waste Processing Exhaust Fans
- 11 AND 12 Access Control Area Exhaust Fans

The Fuel Handling Procedure FH-305 requires fuel handling to be suspended upon a change in Ventilation. (From FH-305 pg 14)

**5.6 Ventilation**

Fuel movement will be suspended in the event of the loss of one of two operating air supply fans, or a change in Auxiliary Building ventilation lineup with a single fan operating. **[B-152]**

Also The Refuel Machine Procedure OI-25C requires the performance of OI-22D Appendix C Checklist before use of the Refuel machine is permitted. (From OI-25C pg. 10)

c. IF Refueling Operations will begin,
**THEN PERFORM** the following:

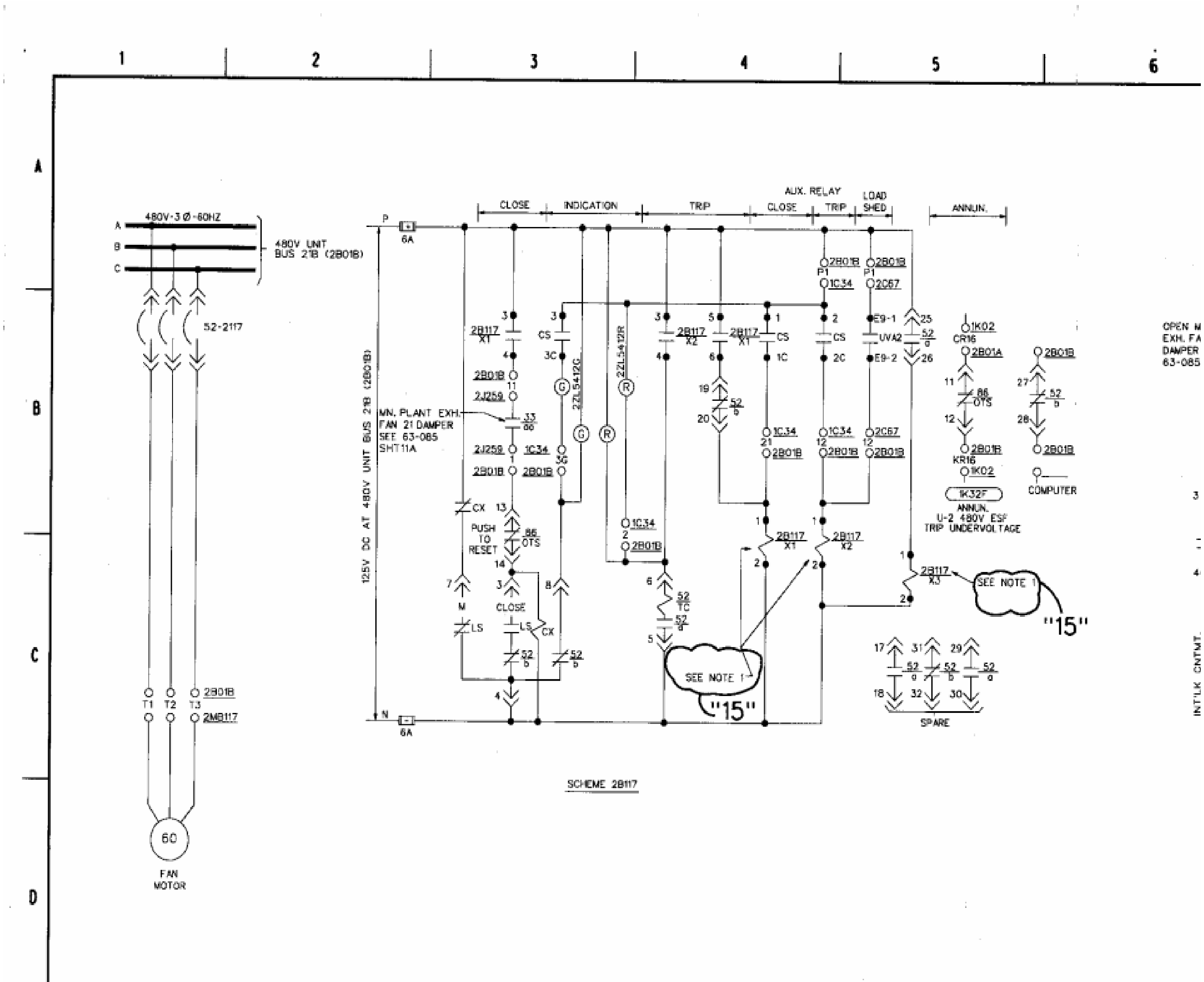
- **ENSURE** APPENDIX C, REFUELING OPERATIONS CHECKLIST is complete. **[B0408]** (N/A if performed as part of APPENDIX A)
- **ENSURE** OI-22D, FUEL HANDLING AREA VENTILATION SYSTEM APPENDIX C, VENTILATION WALKDOWN CHECKLIST is complete. **[B0408]**

The OI-22D Appendix C checklist requires one Aux bldg supply fan running per unit. (From OI-22D Appendix C. pg 2)

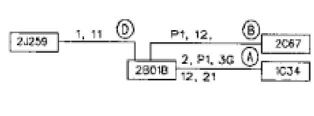
## VENTILATION WALKDOWN CHECKLIST [B0408]

D. Spent Fuel Pool Area Ventilation Systems and Components
(N/A if **NOT** moving recently irradiated fuel assemblies in the Auxiliary Building)

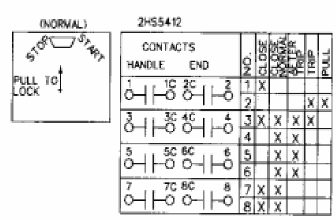
- **VERIFY** 11 or 12 Main Vent Exhaust Fan in operation ____
- Locally **VERIFY** two Auxiliary Building Supply Fans are running (one per unit) ____
- Locally **VERIFY** Spent Fuel Pool Exhaust Fan running ____
- **VERIFY** standby Spent Fuel Pool Exhaust Fan **NOT** rotating backwards ____
- **VERIFY** Spent Fuel Pool Supply Fans **NOT** in operation ____
- **VERIFY** Spent Fuel Pool Exhaust Ventilation filters in service:
  - 0-HS-5416 in FILTER position at 1C34 ____
  - 0-PDI-5417 reads greater than or equal to .9" $H_2O$. ____
  - **SUM** the readings on 0-PDI-5417 and 0-PDIS-5418 to ensure the combined SFP Roughing, HEPA, and Charcoal Filter delta p are less than 4.0" $H_20$. ____



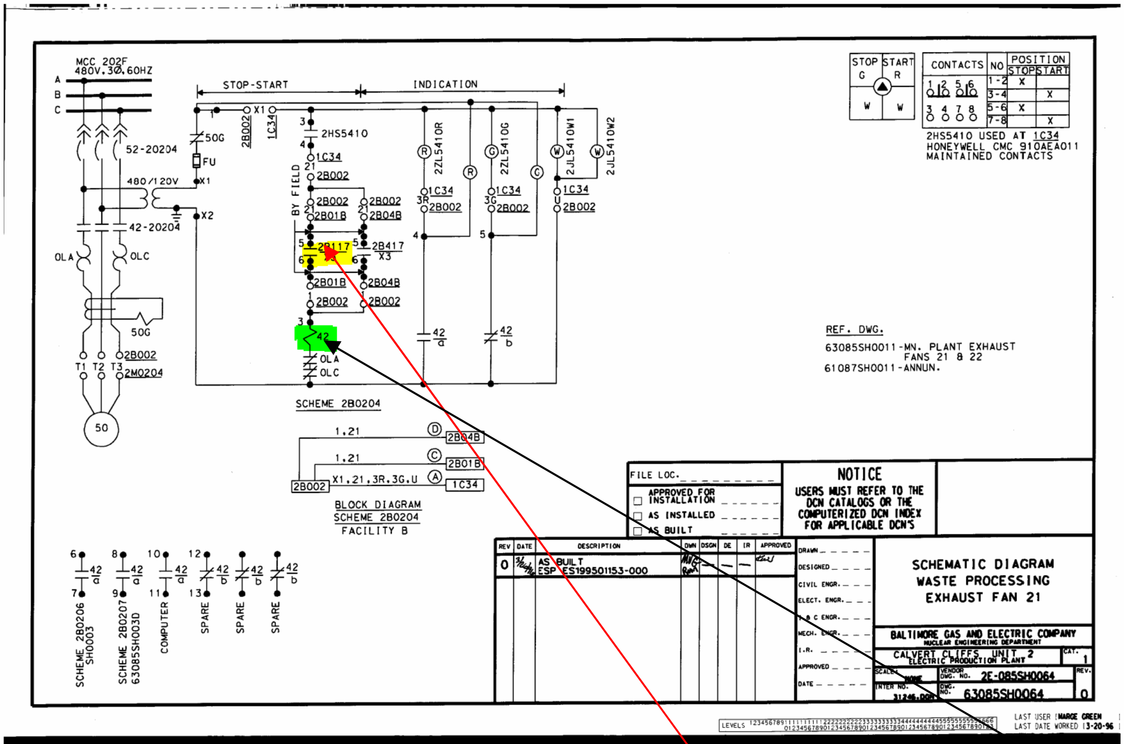

| SCHEME NO. | DESRIPTION | BKR. NO. | LOCATIONS | | | INDICATING LAMPS | RELAY | CONTR. SW. | FAC. | CONTACTS | AUX. RELAY | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 2B117 | MAIN EXHAUST FAN 21 | 52-2117 | 2B01B | 2J259 | 2C67 | 2ZL5412R & G | UVA2 | 2-HS-5412 | A | E9-1, E9-2 | 2B117 X1 | 2B117 X2 | 2B117 X3 |
| 2B417 | MAIN EXHAUST FAN 22 | 52-2417 | 2B04B | 2J260 | 2C68 | 2ZL5413R & G | UVB2 | 2-HS-5413 | B | E1-1, E1-2 | 2B417 X1 | 2B417 X2 | 2B417 X3 |



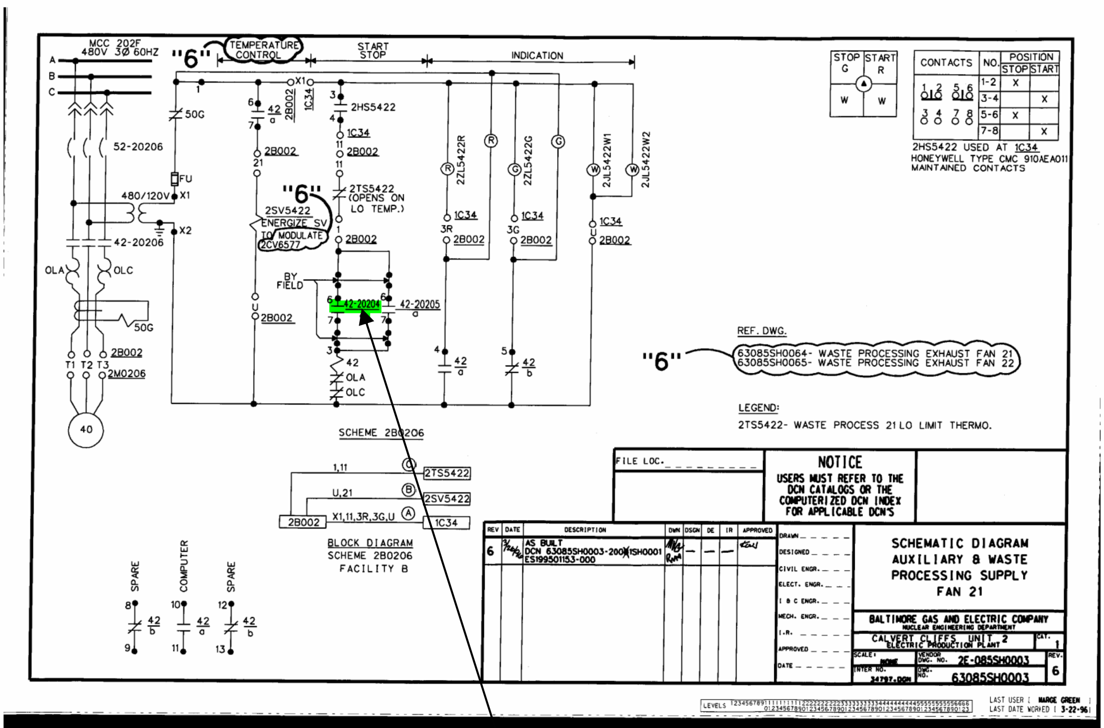

BLOCK DIAGRAM
SCHEME 2B117
FACILITY A




CS USED AT 1C34
SPRING RETURN TO NORMAL
G.E. TYPE SBM MODEL NO. 10AA122




Aux Relay 2B117 de-energizing will Open contact 2B117 , which will de-energize 42 relay
X3 X3




The 42 relay de-energizing will cause the 42-20204 contact to open causing the Auxiliary Building supply fan to trip a

**Regrade Request**

SRO Question #24 should be deleted due to no correct answer.

**A.** Refueling pool level decreases--**Is Correct**. The Main Exhaust Fan tripping would cause Containment Purge to secure. This would cause containment pressure to rise slightly, with the transfer tube gate valve open, refueling pool level will decrease accordingly due to the differential pressure between the SFP area and containment. (SFP is maintained at a slight negative pressure) Continue refueling operations is **Incorrect** due to the above justification.

**B**. (a) Containment pressure rises 1 to 2 PSIG, (b) Initiate additional containment cooling --**Incorrect** The Main Exhaust Fan tripping would cause Containment Purge to secure. This would cause containment pressure to rise slightly, Containment pressure rises 1 to 2 PSIG--**incorrect**, containment pressure will change only a few tenths of a pound per square inch and there is no procedural guidance to increase containment cooling.

**C.** Area radiation monitors (RE-5316A-D) indicate higher--**incorrect**, the area monitors' indications do not change if Containment Purge is lost.

**D.** Refueling pool level increases-- **incorrect**. The Main Exhaust Fan tripping would cause Containment Purge to secure. This would cause containment pressure to rise slightly, with the transfer tube gate valve open, refueling pool level will decrease accordingly due to the differential pressure between the SFP area and containment. (SFP is maintained at a slight negative pressure)

**Question Statistics**
Question SRO 24 was missed by 5 candidates. Two candidates selected A, one selected B, one selected C and three selected D.

**Post Examination Review References**

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|---|---|
| FH-305 | CORE ALTERATIONS |
| OI-25C | MAIN EXHAUST FAN SYSTEM |
| OI-22D | FUEL HANDLING AREA VENTILATION SYSTEM |
| DWG No.63085SH0003 | SCHEMATIC DIAGRAM AUXILIARY & WASTE PROCESSING SUPPLY FAN 21 |
| DWG No.63085SH0064 | SCHEMATIC DIAGRAM WASTE PROCESSING EXHAUST FAN 21 |
| DWG No.63085-D SH11 | SCHEMATIC DIAGRAM HEATING AND VENTILATIONMAIN PLANT EXHAUST FANS 21 & 22 |

**NRC Response**: The licensee contends that there is no correct response provided for the conditions that are contained in the question. During the examination development process, it was not anticipated that the loss of the main exhaust fan due to a loss of power would also result in the loss of the auxiliary building and waste processing supply fans. Therefore with no auxiliary building supply fans running, licensee procedures (Fuel Handling Procedure FH-305, Refuel Machine Procedure OI-25C and Fuel Handling Area Ventilation System OI-22D) direct that fuel movement be suspended. Suspending refueling operations was not provided in any of the choices for this question. Therefore, there is no correct answer to this question.

**Conclusion**: The NRC agrees with the licensee's recommendation to delete this question because there is no correct answer. The question will be deleted.