January 21, 2000

NOTE TO: NRC Document Control Desk
Mail Stop 0-5-D-24

Beverly L Michael

FROM: Beverly Michael, Licensing Assistant, Operator Licensing and Human Performance Branch, Division of Reactor Safety, Region II

SUBJECT: OPERATOR LICENSING EXAMINATIONS ADMINISTERED AT THE EDWIN I. HATCH NUCLEAR PLANT, DOCKET NOS. 50-321 AND 50-366 -

During the period October 29 and November 1 - 4, 1999, Operator Licensing Examinations were administered at the referenced facility. Attached, you will find the following information for processing through NUDOCS and distribution to the NRC staff, including the NRC PDR:

Item #1 - a) Facility submitted outline and initial exam submittal, designated for distribution under RIDS Code A070.

b) As given operating examination, designated for distribution under RIDS Code A070.

Item #2 - Examination Report with the as given written examination attached, designated for distribution under RIDS Code IE42.

Attachments: As stated

Examination report with the as given written examination attached

designated for distribution under RIDS Code IE42

DISTRIBTION CODE
IE42

December 6, 1999

Southern Nuclear Operating Company, Inc.
ATTN: Mr. H. L. Sumner, Jr.
Vice President
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: NRC EXAMINATION REPORT 50-321/99-301, 50-366/99-301

Dear Mr. Sumner:

On November 5, 1999, the Nuclear Regulatory Commission (NRC) completed administering operator licensing examinations to employees of your company who had applied for licenses to operate your Edwin I. Hatch Nuclear Plant Units 1 and 2. The enclosed report presents the results of that examination.

All seven senior reactor operator applicants who were administered the written examination and operating test, passed the examination representing a 100 percent pass rate. A Simulation Facility Report is included in this report as Enclosure 2. A copy of the written examination questions and answer key as noted in Enclosure 3, was retained by your facility following administration.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

Original signed by
Harold O. Christensen

Harold O. Christensen, Chief
Operator Licensing and Human
Performance Branch
Division of Reactor Safety

Docket Nos.: 50-321, 50-366
License Nos.: DPR-57, NPF-17

Enclosures:
1. Report Details
2. Simulation Facility Report
3. Written Examination and Answer Key (SRO)
   (Document Control Desk Only)

cc w/encls: (See page 2)

DISTRIBTION CODE
IE42

cc w/encls:
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Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

P. H. Wells
General Manager, Plant Hatch
Southern Nuclear Operating Company, Inc.
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D. M. Crowe
Manager Licensing - Hatch
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Distribution w/encls:
L. Olshan, NRR
PUBLIC

*FOR PREVIOUS CONCURRENCE SEE ATTACHED COPY

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRP	RII:DRS		
SIGNATURE							
NAME	*CPayne:pd	*MErnstes	RAiello	*PSkinner	HChristensen		
DATE	12/ /99	12/ /99	12/ 1 /99	12/ /99	12/ 1 /99	12/ /99	12/ /99
COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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L. Olshan, NRR
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OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:DRP		
SIGNATURE	[signature]	[signature] for			[signature]		
NAME	CPayne:pd	MErnstes	RAiello	HChristensen	PSkinner		
DATE	11/30/99	11/30/99	11/ /99	11/ /99	11/30/99	11/ /99	11/ /99
COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: P:\Hatch992.WPD

Rec'd via E-mail (Attached)

From: Michael Ernstes
To: Charlie Payne
Date: Tue, Nov 30, 1999 12:01 PM
Subject: Re: HATCH EXAM REPORT

I CONCUR!

>>> Charlie Payne 11/30 11:53 AM >>>
Mike,

Subject report attached. Let me know if you concur.

Thanks,
Charlie...

NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: License Nos.:	50-321, 50-366 DPR-57, NPF-17
Report No.:	50-321/99-301, 50-366/99-301
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Edwin I. Hatch Nuclear Plant Units 1 and 2
Location:	11030 Hatch Parkway North Baxley, GA 31513
Dates:	Written Examination - October 29, 1999 Operating Tests - November 1 - 4, 1999
Examiners:	C. Payne, Senior Operations Engineer M. Ernstes, Senior Operations Engineer R. Aiello, Senior Operations Engineer
Approved by:	H. Christensen, Chief Operator Licensing and Human Performance Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Edwin I. Hatch Nuclear Plant Units 1 and 2
NRC Examination Report 50-321/99-301, 50-366/99-301

This report documents the results of cooperative effort between the licensee and regional examiners to develop, validate and administer operator licensing initial examinations in accordance with the guidance of Examination Standards, NUREG-1021, Revision 8. This examination implemented the operator licensing requirements of 10 CFR §55.41, §55.43, and §55.45.

Seven senior reactor operator candidates were administered the final, approved written examination and operating test. The licensee and NRC administered the written examination on October 29, 1999. The NRC administered the operating tests during the week of November 1, 1999.

Operations

- All seven senior reactor operator (SRO) candidates passed the examination and were issued SRO licenses. (Section O5.1; [POS-3B])

- In general, candidate performance on the written examination and the operating test was very good. Candidate knowledge, skill and ability weaknesses were few in number and narrow in scope. Each candidate was well prepared for the operator licensing examination. (Section O5.1; [POS-3B])

Report Details

Summary of Plant Status

During the period of the examinations, both units operated at 100 percent power.

I. Operations

O5 Operator Training and Qualifications

O5.1 Initial Operator Licensing Examinations

a. Examination Scope

NRC examiners administered regular, announced operator licensing examinations developed by the licensee and NRC in accordance with the guidelines of the Operator Licensing Examination Standards (ES) for Power Reactors, NUREG-1021, Revision 8 during the period November 1-4, 1999. The written examination was administered by the licensee and the NRC on October 29, 1999. Seven senior reactor operator (SRO) license applicants received the written examination and operating test. The examiners reviewed the results of the written examination and evaluated the candidates' compliance with and use of plant procedures during the simulator scenarios and job performance measures (JPMs).

b. Observations and Findings

The NRC developed the SRO written examination which the licensee reviewed and validated. The licensee developed one administrative test set (three JPMs and four questions), one plant systems test set (10 JPMs), and four simulator scenarios which the NRC examiners reviewed. The examiners validated these test items during a site preparation visit the week of October 4, 1999.

Examination Results and Conclusions

All seven candidates passed the examination. The average score of the written examination was 88 percent (80 percent was required for passing) which indicated that the candidates were well prepared. The licensee did not submit any formal post-examination comments on the written examination.

The NRC conducted a post-examination item analysis of the written examination. The examiners identified only four questions (#s 18, 68, 81 and 90) which were answered incorrectly by 50 percent or more of the candidates. The examiners reaffirmed that each question tested valid knowledge and ability areas. One question (# 18) was missed by five of the seven (72 percent) candidates and may indicate a generic weakness in the candidates' understanding of EOP overrides. The examiners noted that this question could also be improved with additional clarifying information. Wording changes were incorporated into the NRC question bank for use on future examinations. The examiners evaluated the three remaining questions (#s 68, 81 and 90) and determined that they were adequate as written. These questions were missed by four of the seven (57 percent) candidates and may represent a more limited knowledge weakness by the affected candidates' in the specific areas tested by these questions.

The examiners noted one generic candidate performance weakness during the plant walkthrough examinations. Four of seven (57 percent) candidates had difficulty in properly responding to a loss of lubricating oil condition while an Emergency Diesel Generator (EDG) was being operated in the TEST mode. Each apparently did not understand that the EDG could not be tripped (by taking the diesel START switch to the STOP position) unless the EDG output breaker was open.

Details of these and other discrepancies were described in each individual's examination report, Form ES-303-1, "Operator Licensing Examination Report," which have been forwarded under separate cover to the Training Manager. This will enable you to evaluate the weaknesses and provide appropriate feedback and/or remedial training as necessary. This information was also provided so you may evaluate whether a training program oversight in the above areas may be present.

c. Conclusions

All seven candidates were issued SRO licenses. The examiners concluded that overall candidate performance on the written examination and the operating test was very good. Candidate knowledge, skill and ability weaknesses were few in number and narrow in scope. Each candidate was well prepared for the operator licensing examination.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the examination results to members of licensee management at the conclusion of the examination on November 5, 1999. The licensee acknowledged the findings presented. The licensee commented that the cooperative effort in developing these examinations worked well and was considered a positive experience.

The examiners asked the licensee whether any materials used during the examination should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

*R. Belcher, Plant Instructor
*R. Grantham, Supervisor, Operations Training
*J. Lewis, Training and Emergency Preparedness Manager
*D. Madison, Operations
*C. Moore, Assistant General Manager, Plant Support
*R. Smith, Plant Instructor
*S. Tipps, Nuclear Safety Compliance Manager
*K. Underwood, Safety Audit and Engineering Review

NRC

*T. Fredette, Resident Inspector
J. Munday, Senior Resident Inspector

*Attended Exit Meeting

INSPECTION PROCEDURES USED

NUREG-1021, Rev. 8: Operator Licensing Examination Standards for Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedures
ES	Examination Standard
JPM	Job Performance Measure
NRC	Nuclear Regulatory Commission
SRO	Senior Reactor Operator

SIMULATION FACILITY REPORT

Facility Licensee: E. I. Hatch Nuclear Plant Unit 1 and Unit 2

Facility Docket Nos.: 50-321 and 50-366

Operating Tests Administered on: November 1 through 4, 1999

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

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

<u>ITEM</u>	<u>DESCRIPTION</u>
HPCI Auxiliary Oil Pump	The applicants were administered a JPM where the HPCI System had automatically started due to a spurious initiation signal. The initiation signal had been reset and the applicant was directed to shut down the system per the normal operating procedure. As part of the system shutdown, the operator confirms that the Auxiliary Oil Pump automatically starts prior to turbine speed decreases below 1500 rpm. This occurred as expected. Later in the system restoration process, part of the system realignment for automatic start requires the operator to stop the running Aux Oil Pump. However, it was found that the pump had shut itself down prior to any operator action. The licensee indicated this system response was incorrect and a simulator deficiency was entered into their simulator repair tracking system. The problem was manually overridden by the simulator operator for the remainder of the testing on this task.

Public Copy
(Questions, Answers,
References)

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: II
Date: October 29, 1999	Facility/Unit: E.I. Hatch 1 & 2
License Level: SRO	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	________	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

DO NOT TURN PAGE UNTIL TOLD SO.

HATCHSRO.TST Version: 0

Name ______________________________

1. [a] [b] [c] [d] ____	35. [a] [b] [c] [d] ____	69. [a] [b] [c] [d] ____
2. [a] [b] [c] [d] ____	36. [a] [b] [c] [d] ____	70. [a] [b] [c] [d] ____
3. [a] [b] [c] [d] ____	37. [a] [b] [c] [d] ____	71. [a] [b] [c] [d] ____
4. [a] [b] [c] [d] ____	38. [a] [b] [c] [d] ____	72. [a] [b] [c] [d] ____
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8. [a] [b] [c] [d] ____	42. [a] [b] [c] [d] ____	76. [a] [b] [c] [d] ____
9. [a] [b] [c] [d] ____	43. [a] [b] [c] [d] ____	77. [a] [b] [c] [d] ____
10. [a] [b] [c] [d] ____	44. [a] [b] [c] [d] ____	78. [a] [b] [c] [d] ____
11. [a] [b] [c] [d] ____	45. [a] [b] [c] [d] ____	79. [a] [b] [c] [d] ____
12. [a] [b] [c] [d] ____	46. [a] [b] [c] [d] ____	80. [a] [b] [c] [d] ____
13. [a] [b] [c] [d] ____	47. [a] [b] [c] [d] ____	81. [a] [b] [c] [d] ____
14. [a] [b] [c] [d] ____	48. [a] [b] [c] [d] ____	82. [a] [b] [c] [d] ____
15. [a] [b] [c] [d] ____	49. [a] [b] [c] [d] ____	83. [a] [b] [c] [d] ____
16. [a] [b] [c] [d] ____	50. [a] [b] [c] [d] ____	84. [a] [b] [c] [d] ____
17. [a] [b] [c] [d] ____	51. [a] [b] [c] [d] ____	85. [a] [b] [c] [d] ____
18. [a] [b] [c] [d] ____	52. [a] [b] [c] [d] ____	86. [a] [b] [c] [d] ____
19. [a] [b] [c] [d] ____	53. [a] [b] [c] [d] ____	87. [a] [b] [c] [d] ____
20. [a] [b] [c] [d] ____	54. [a] [b] [c] [d] ____	88. [a] [b] [c] [d] ____
21. [a] [b] [c] [d] ____	55. [a] [b] [c] [d] ____	89. [a] [b] [c] [d] ____
22. [a] [b] [c] [d] ____	56. [a] [b] [c] [d] ____	90. [a] [b] [c] [d] ____
23. [a] [b] [c] [d] ____	57. [a] [b] [c] [d] ____	91. [a] [b] [c] [d] ____
24. [a] [b] [c] [d] ____	58. [a] [b] [c] [d] ____	92. [a] [b] [c] [d] ____
25. [a] [b] [c] [d] ____	59. [a] [b] [c] [d] ____	93. [a] [b] [c] [d] ____
26. [a] [b] [c] [d] ____	60. [a] [b] [c] [d] ____	94. [a] [b] [c] [d] ____
27. [a] [b] [c] [d] ____	61. [a] [b] [c] [d] ____	95. [a] [b] [c] [d] ____
28. [a] [b] [c] [d] ____	62. [a] [b] [c] [d] ____	96. [a] [b] [c] [d] ____
29. [a] [b] [c] [d] ____	63. [a] [b] [c] [d] ____	97. [a] [b] [c] [d] ____
30. [a] [b] [c] [d] ____	64. [a] [b] [c] [d] ____	98. [a] [b] [c] [d] ____
31. [a] [b] [c] [d] ____	65. [a] [b] [c] [d] ____	99. [a] [b] [c] [d] ____
32. [a] [b] [c] [d] ____	66. [a] [b] [c] [d] ____	100. [a] [b] [c] [d] ____
33. [a] [b] [c] [d] ____	67. [a] [b] [c] [d] ____	
34. [a] [b] [c] [d] ____	68. [a] [b] [c] [d] ____	

<u>WRITTEN EXAMINATION GUIDELINES</u>

1. ***[Read Verbatim]*** After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

2. To pass the examination, you must achieve a grade of 80.00 percent or greater; grades will not be rounded up to achieve a passing score. Every question is worth one point.

3. The time limit for completing the examination is five hours.

4. You may bring pens, pencils, and calculators into the examination room. Use black ink to ensure legible copies; dark pencil should be used only if necessary to facilitate machine grading.

5. Print your name in the blank provided on the examination cover sheet and the answer sheet. You may be asked to provide the examiner with some form of positive identification.

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you are using ink and decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If you have any questions concerning the intent or the initial conditions of a question, do *not* hesitate asking them before answering the question. Ask questions of the NRC examiner or the designated facility instructor *only.* When answering a question, do *not* make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question.

8. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

10. After you have turned in your examination, leave the examination area as defined by the proctor or NRC examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

11. Do you have any questions?

Hatch Nuclear Plant
Senior Reactor Operator Written Examination

October 29,1999

Test: HATCHSRO.TST

Date: Wednesday, October 27, 1999

Version: 0

DO NOT TURN PAGE UNTIL TOLD TO DO SO.

1. The Unit 2 Station Service 4160V buses are operating in a normal lineup with the unit at MOP. The normal supply breaker to 4160V bus "2C" inadvertantly trips. There is no fault on the bus. Which one of the following equipment responses should occur?

   a. The normal supply breaker for that bus auto recloses after 5 seconds since there is not a fault on the bus.
   b. The bus fast transfers and the alternate supply breaker auto closes to maintain power to the bus.
   c. The alternate supply breaker for that bus closes to re-energize the bus and EDG "2C" receives an auto start signal.
   ✓d. The alternate supply breaker for that bus will not close and the bus remains de-energized.

   Bank question (slightly modified)
   LT-LP-02702-03, p. 13

2. Unit 1 is operating at 50% power when the APRM "B" fails upscale. Before APRM "B" can be bypassed, the RPS power to the "A" two-out-of-four Logic Module is lost. Which one of the following describes the resulting status of the RPS system?

   a. Both RPS "A" and RPS "B" scram relays are energized.
   ✓b. RPS "A" scram relays are deenergized and RPS "B" scram relays are energized.
   c. RPS "A" scram relays are energized and RPS "B" scram relays are deenergized.
   d. Both RPS "A" and RPS "B" scram relays are deenergized.

   Bank question (reworded slightly)
   SI-LP-01203-00, pp. 8, 10-11, 26-27

3. The following plant conditions have existed on Unit 2 for the last 5 minutes:

   A LOSP has occurred
   All 4160 V emergency buses are deenergized.
   No EDGs are currently running.
   HPCI and RCIC are isolated.
   RPV level is -110 inches and steady.
   Drywell pressure is 4.2 psig.
   Inhibit switches are in NORMAL.

   Which one of the following describes the status of the ADS?

   a. ADS has initiated and 7 ADS valves should be open.
   b. ADS will initiate in approximately 8 minutes when the 13 minute timer times out.
   ✓c. ADS will initiate immediately after AC power is restored and a low pressure ECCS pump is started.
   d. ADS will initiate 2 minutes after AC power is restored and a low pressure ECCS pump is started.

   '97 NRC exam, Q# 10
   SI-LP-03801-00, pp. 4-7

4. Which one of the following is TRUE with the Emergency Transfer Switches at the Remote Shutdown Panels (RSDPs) in the NORM position?

   a. Neither Unit can control equipment from the RSDPs.

   ✓b. Unit 1 can control equipment from the RSDPs but Unit 2 can not.

   c. Unit 2 can control equipment from the RSDPs but Unit 1 can not.

   d. Both units can control equipment from the RSDPs.

   New question
   SI-LP-05201-00, p. 8

5. Preparations are presently being made to refuel the Unit 1 reactor. The following plant conditions exist:

Reactor Mode Switch position:	SHUTDOWN
Reactor pressure:	0 psig
Reactor temperature:	180°F
RPV head closure bolts:	Detensioning is 10% complete
Control rod position:	All rods in

Based on the above conditions, the reactor would be in which one of the following Modes?

a. Mode 2
b. Mode 3
c. Mode 4
✓d. Mode 5

SRO only
'97 NRC exam, Q#84 (modified)
Unit 1 TS, p. 1.1-7, Table 1.1-1
LT-LP-30005-04, p. 6

6. Fuel movement is in progress on Unit 2 with the following plant conditions:

	Mode Switch Position	Coolant Temperature	Reactor Power
Unit 1 Unit 2	Run Refuel	545°F 128°F	80% 0%

Which one of the following is the minimum on-site shift staffing required by the Unit 2 Technical Specifications?

a. SRO 1 + 1 for Fuel Handling
RO 2
PEO 3
STA 0
b. SRO 1 + 1 for Fuel Handling
RO 2
PEO 3
STA 1
c. SRO 2 + 1 for Fuel Handling
RO 2
PEO 3
STA 1
✓d. SRO 2 + 1 for Fuel Handling
RO 3
PEO 3
STA 1

SRO only
'93 NRC exam, Q# 87 (updated for ITS)
Unit 2 TS, p. 5.0-1 - 5.0-4
30AC-OPS-003-0S, p. 35
10 CFR 50.54(m)(2)(i)
LT-ST-30003-05, p. 8

7. Which one of the following documents is required to be reviewed by the Shift Supervisor <u>prior</u> to assuming shift per 31GO-OPS-007-0S, "*Shift Logs and Relief of Personnel*"?

   ✓a. Control Room Log
   b. Annunciator Control Log
   c. Temporary Modification Log
   d. RAS Log

   SRO only
   Bank question (modified slightly)
   31GO-OPS-007-0S, p. 3
   LT-LP-30004-04, p. 32

8. While operating mid-cycle on Unit 2, chemistry reports that the Zinc Injection System has failed and repair parts will not be available for 6 months. Which one of the following is the consequence of operating the reactor without the injection of zinc?

   a. The potential for Main Condenser tube leaks is increased.
   b. The potential for Intergranular Stress Corrosion Cracking is increased.
   ✓c. Dose rates in the drywell during outages will increase due to more Cobalt-60 plating out on primary system components.
   d. Dose rates out the Main Stack will increase due to less effective Iodine-133 scrubbing in the Off-gas system.

   New question
   LT-LP-07301-04, pp. 8, 34-35

9. According to the current Facility Operating License No. DPR-57, which one of the following is the maximum power level authorized by NRC for Unit 1?

   a. 897 MWe
   b. 943 MWe
   ✓c. 2763 MWt
   d. 2816 MWt

   New question
   Facility Operating License No. DPR-57
   Unit 1 Tech. Spec., Sect. 1.1, p. 1.1-5
   LT-LP-30005-04, p. 7 (note: U1 data in LP is incorrect, see license)

10. Unit 1 is operating at 80% power. "1A" CRD Pump is in service. The operators observe the following indications:

Charging water pressure:	Low
Cooling water flow:	Low
Drive water flow:	Low
Cooling water dP:	Low
Drive water dP:	Low
CRD Mechanism temperatures:	Rising
Recirc Pump seal temperatures:	Rising

Which one of the following CRD component problems has caused these abnormal conditions? (References included)

✓a. The drive water filter is plugged.
b. The flow control valve has failed closed.
c. The cooling water control valve has failed closed.
d. The drive water pressure control valve has closed.

<u>Provide a copy of Fig. 12 to LP-00101-00 as a reference.</u>

New question
SI-LP-00101-00, p. 8, 35

11. Unit 2 was operating at MOP when a reactor scram occurred. The following plant conditions exist:

Main turbine is tripped.
Main generator PCBs are closed.
Position indication for DC powered RCIC valves is out.
CORE SPRAY SYS I LOGIC POWER FAILURE annunciator is lit.

Which one of the following is the likely cause of this event?

✓a. Loss of 125/250 VDC Switchgear A (2R22-S016)
b. Loss of 125/250 VDC Switchgear B (2R22-S017)
c. Loss of 125 VDC Cabinet B (2R25-S002)
d. Loss of 125 VDC Cabinet C (2R25-S003)

'97 NRC exam, Q# 33 (modified)
34AB-R22-001-2S, p. 8
LT-LP-02704-03, pp. 31, 32

12. Unit 1 is operating at MOP with the "1A" EDG in TEST and paralleled to the "1E" 4160 VAC bus. While testing the EDG, a LOCA occurs. Which one of the following describes the electrical plant response to this event?

   a. The EDG comes out of TEST and all 4160 VAC station service buses deenergize.
   b. The EDG remains paralled to the "1E" bus and all 4160 VAC station service buses deenergize.
   ✓c. The EDG comes out of TEST and all 4160 VAC station service buses transfer to alternate supply.
   d. The EDG remains paralled to the "1E" bus and all 4160 VAC station service buses transfer to alternate supply.

   '97 NRC exam, Q# 22 (modified)
   LT-LP-02702-03, p. 9-17

13. Unit 1 was operating at 35% power when a loss of offsite power transient occurred. If the 600 VAC Nonessential Load Lockout protection failed to function, which one of the following is a possible a consequence of this failure?

✓a. The Emergency Diesel Generators could be overloaded.
b. 4160V buses "1C" & "1D" could experience an overcurrent condition.
c. The Emergency Diesel Generators could trip due to underfrequency.
d. Essential loads on 600V buses "1C" & "1D" could fail to automatically restart.

Bank question (reworded and reordered)
LT-LP-02703-03, p. 19

14. Which one of the following Emergency Director responsibilities may be delegated to another individual?

   a. The decision to request federal assistance.
   b. The decision to notify offsite emergency response agencies.
   ✓c. The decision to evaluate and implement protective actions on the plant site.
   d. The decision to declare, escalate, or downgrade emergency classifications.

   SRO only
   New question
   73EP-EIP-004-0S, p. 2
   EP-LP-20101-00, pp. 14-15

15. Which one of the following is the basis for initiating drywell sprays before the bulk drywell temperature reaches the drywell design temperature limit?

✓a. To ensure that equipment within the drywell will operate when required.
b. To prevent increased degradation of structural concrete and release of hydrogen to the drywell.
c. To ensure that the capacity of the suppression chamber - drywell vacuum breakers is not exceeded.
d. To maintain the equipment qualification of the drywell valves above the 185' elevation, capable of removing the full decay heat load following a LOCA.

New question
EOP flowchart PC-1, G-6
LR-LP-20310-05, p. 59

16. A Site Area Emergency has been declared on Unit 2 with all on-site facilities activated. Per the 73EP-EIP-004-0S, "*Duties of Emergency Director*," which one of the following personnel is the lowest level of management that may assume the Emergency Director duties from the SOS? (Assume all personnel are performing their normal duties)

a. Manager of NSAC, as the EOF Manager.
b. Manager of Operations, as the TSC Manager.
✓c. Plant Operations - Assistant General Manager, in the TSC.
d. Nuclear Plant - General Manager, in the EOF.

SRO only
New question
73EP-EIP-004-0S, p. 4, step 7.2.1
EP-LP-20101-00, p. 13

17. Which one of the following represents the major threat to the public during a severe reactor accident with substantial core damage?

   a. Gamma radiation that is being emitted directly from the damaged fuel.
   ✓b. Radioactive contamination and radiation shine from the release plume.
   c. Explosion that occurs due to buildup of hydrogen over the course of the accident.
   d. Steam explosion that occurs when the core melts and relocates to the containment base mat.

   Bank question (reworded and modified slightly)
   LT-LP-20018-01, p. 9

18. An ATWS has occurred on Unit 2. The SS is currently at the WAIT UNTIL block at coordinate H-8 on EOP flowchart CP-3. The following plant conditions exist:

SBLC tank level:	30%
Reactor power:	5%
Reactor water level:	-140 inches, steady
HP injection systems available:	RCIC, boron and CRD

If reactor power starts to slowly increase, which one of the following actions is required to be performed per CP-3? (References included)

✓a. Evaluate the reactor power increase per the override at coordinate C-2 only.
b. Determine that no CP-3 flowchart overrides are required to be evaluated under current plant conditions.
c. Evaluate the reactor power increase per the override at coordinate F-7 and return to the override at coordinate C-2 to continue the evaluation.
d. Determine that reactor water level can not be restored and Emergency Depressurization is required. Go to Point L on flowchart CP-3.

Provide a copy of the CP-3 EOP flowchart as a reference.

SRO only
New question
EOP Flowchart CP-3
LR-LP-20303-03, p. 11-12

19. While reviewing a procedure that is required to be completed before the end of the current shift, the SS notices a step requiring the use of a gauge which is broken. Another gauge is available in the system and the SS has confirmed it will operationally function as a substitute. At a minimum, which one of the following actions must be done to perform the procedure? The SS should:

   a. Make a pen and ink change to the procedure.
   b. Make a pen and ink change to the procedure with SOS concurrence.
   ✓c. Make a SRO change to the procedure.
   d. Make a permanent change to the procedure obtaining manager approval prior to use.

   SRO only
   New question
   10AC-MGR-003-0S, pp. 9-10
   LT-LP-30004-04, pp. 15-17

20. The automatic scram signals on Unit 2 have been overridden per the EOPs during an ATWS. Per 40AC-ENG-018-0S, "*Temporary Modification Control,*" which one of the following actions must the crew perform for controlling these jumpers?

✓a. No action is required for these conditions.
b. Complete the temporary modification form only.
c. Complete and attach temporary modification tags to the jumpers only.
d. Complete the temporary modification form and attach temporary modification tags.

SRO only
Bank question (reworded to eliminate teaching in correct answer)
40AC-ENG-018-0S
LT-LP-30004-04, p. 41

21. Unit 2 is in a refueling outage with a full core offload in progress. A fuel bundle is being transfered from the core to the fuel pool when the control room operator reports that reactor cavity water level is decreasing. Per 34AB-G41-002-2S, "*Decreasing Rx Well/Fuel Pool Water Level,*" which one of the following actions should the refueling SRO direct the bridge operator to perform?

   a. Return the fuel bundle to any in-core location that is available.
   ✓b. Move the fuel bundle to any fuel storage rack in the fuel pool.
   c. Move the fuel bundle to the fuel pool and lower it as deep into the pool as possible.
   d. Move the fuel bundle to the transfer canal and lower it as deep into the canal as possible.

   Bank question (modified slightly)
   34AB-G41-002-2S
   LT-LP-04502-03, p. 36

22. An event caused the "2G" 4160 VAC bus to be de-energized. The "2C" Emergency Diesel Generator is not supplying bus "2G". The cause of the electrical failure has been found and corrected. Which one of the following represents the necessary approval(s) required to reset the LOSP lock-out relay and restore normal power to the bus?

    a. Shift Supervisor only.
    b. Maintenance Supervisor (Electrical) only.
    ✓c. Shift Supervisor and Supervisor Engineering Support.
    d. Unit Superintendent and Maintenance Supervisor (Electrical).

    Bank question (reworded slightly)
    30AC-OPS-003-0S, sec. 8.5.1, p. 15
    LT-LP-30007-01, p. 27

23. Which one of the following conditions will <u>directly</u> result in an automatic start of both diesel fire pumps?

✓a. A loss of offsite power.
b. A loss of instrument air.
c. A fire alarm on the XL3 Master Panel.
d. A sustained low fire main pressure of 110 psig.

Bank question (distractors rearranged, "d" changed)
34SO-X43-001-2S
LT-LP-03601-03, p. 21

24. Core offload is in progress. All control rods are fully inserted and the Mode Switch is in REFUEL. A fuel assembly has just been placed in the fuel pool and unlatched. The fuel grapple has been raised to "Fully Up" to pass through the cattle chute with the bridge still over the fuel pool location. The next step requires that another fuel assembly be removed from the reactor core and moved to the fuel pool.

Which one of the following will cause a Rod Block as the next step is performed? (Consider each alternative separately)

✓a. While over the reactor, the fuel grapple is loaded.
b. As soon as the bridge begins traveling over the reactor.
c. While passing through the cattle chute, two control rods are selected.
d. As the bridge exits the cattle chute, the frame mounted hoist is lowered.

New question (based on '97 BSEP exam Q# 99)
34SV-F15-001-2S, p. 31
LT-LP-04502-03, p. 51

25. A complete loss of Unit 1 service air has occurred. Which one of the following describes the effect this loss will have on the Fuel Pool Transfer Canal inflatable seals?

a. Pressure will be immediately lost, deflating the inner gate seals only.
b. Pressure will be immediately lost, deflating the outer gate seals only.
✓c. Pressure to the seals will not be immediately lost due to air receivers that are available to automatically supply air pressure to the seals.
d. Pressure to the seals will not be immediately lost due to a backup nitrogen bottle that is available to automatically supply pressure to the seals.

'93 NRC exam, Q# 32
SI-LP-04501-00, p. 12

26. A LOCA has occurred on Unit 1. EOP flowcharts PC-1 & PC-2 are being implemented. Plant conditions are as follows:

Drywell pressure:	5.5 psig
Drywell temperature:	165°F
Torus water level:	280 inches
Drywell [H_2]:	6.4%
Drywell [O_2]:	4.7%
Torus [H_2]:	5.2%
Torus [O_2]:	5.6%
Radioactive release rate:	0.63 mR/hr

Based on the above conditions, which one of the following should not be performed to restore primary containment parameters to acceptable levels?

a. Vent the drywell.
b. Initiate torus sprays.
✓c. Operate the drywell cooling fans.
d. Initiate drywell nitrogen purge flow.

SRO only
Bank question (modified)
EOP Flowcharts PC-1 & PC-2, D-10
LR-LP-20310-05, pp. 68, 86

27. Unit 2 has scrammed due to high drywell pressure. The RHR system is operating in the Drywell Spray mode with the following plant conditions:

Drywell average temperature:	198° and decreasing
Drywell pressure:	1.7 psig
Suppression chamber pressure:	1.7 psig
Suppression pool level:	149 inches and stable
Suppression pool temperature:	93°F and increasing
Reactor water level:	-15 inches and increasing

Which one of the following describes the appropriate action for operation of the Drywell Sprays under these conditions?

a. Drywell Sprays should remain in service until the plant is operating inside the safe region of the Drywell Spray Initiation Limit.
b. Drywell sprays should remain in service until drywell/suppression chamber differential pressure reaches -0.5 psid.
c. Drywell Sprays should be secured because suppression chamber pressure is below 1.85 psig.
✓d. Drywell Sprays should be secured before drywell pressure drops to 0.0 psig.

SRO only
Bank question (modified for EOP changes)
EOP flowchart PC-1, E-9
LR-LP-20310-04, p. 46

28. While performing a HPCI surveillance with the "2A" loop of RHR in suppression pool cooling, a HPCI instrument line breaks resulting in drywell pressure rising to 2.15 psig. Which one of the following describes the status of RHRSW after this event and why? RHRSW is:

a. Still running due to not having a LOSP load shed signal.
b. Tripped and cannot be restarted due to the LOCA load shed logic.
✓c. Tripped initially, but can be restarted by overriding the LOCA signal.
d. Tripped initially and then sequentially tied back onto the Emergency Bus due to the LOCA load shed logic.

Bank question (reworded slightly)
34SO-E11-010-2S
SI-LP-03401-02, pp. 26-27

29. Unit 2 is operating at MOP. Drywell cooling unit/fans B007A & B, B008A, B009A, and B010A are in RUN and operating. The crew receives the following alarms and indications:

DRYWELL COOLING UNIT B007A AIR DISCH TEMP HIGH annunciator is lit
DRYWELL COOLING UNIT B007B AIR DISCH TEMP HIGH annunciator is lit
DRYWELL COOLING UNIT B008A AIR DISCH TEMP HIGH annunciator is lit
DRYWELL COOLING UNIT B009A AIR DISCH TEMP HIGH annunciator is lit
DRYWELL COOLING UNIT B010A AIR DISCH TEMP HIGH annunciator is lit
DRWL CHILLED WTR B006A SAFETY S/D annunciator is lit

Drywell Pressure: 0.88 psig and slowly rising

Drywell Temperatures (from SPDS):

UPPER	MIDDLE	LOWER
158°F 149°F 151°F 147°F	138°F 134°F	125°F 122°F 128°F 132°F

All drywell temperatures are slowly rising.

The SS implements 34AB-T47-001-2S, *"Complete Loss of Drywell Cooling,"*. Based on the above indications, which one of the following actions should the SS perform?

a. Correct reactor water level indications due to high drywell temperature.
✓b. Vent the drywell with CAD to control drywell pressure.
c. Immediately commence a Fast Reactor Shutdown per 34GO-OPS-014-2S.
d. Enter EOP PC-2, *"Primary Containment Control,"* due to high drywell temperature.

SRO only
New question
34AB-T47-001-2S
SI-LP-01304-01, pp. 17-19, 27-34

30. The Unit 2 reactor has scrammed. All PSW pumps are tripped and cannot be restarted. The following plant conditions are noted:

Reactor water level:	-5" and increasing slowly
Reactor pressure:	920 psig, controlled with the bypass valves
Drywell temperature:	339°F and increasing slowly
Drywell pressure:	1.4 psig and steady

Based on the above conditions, which one of the following actions should the operators perform? (References included)

a. Initiate drywell sprays.
b. Commence a controlled cooldown within the cooldown limits.
c. Start all available drywell cooling fans, overriding any automatic trips.
✓d. Emergency depressurize the RPV.

Provide copy of EOP Graph 8, DSIL as a reference.

SRO only
'97 NRC exam, Q# 76 (modified)
LR-LP-20310-05, p. 61

31. Both units are at MOP. Given the following conditions on Unit 1:

All ventilation systems are in a normal line up.
Supply Fan C001A is in RUN and C001B is in STANDBY.
Accessible Area exhaust fan C004A in RUN and C004B is in STANDBY.
Inaccessible Exhaust Fan C007A is in RUN and C007B is in STANDBY.
The accessible area ventilation exhaust radiation monitors (K607A/B) receive a high alarm.

Which one of the following describes the expected automatic response of the Unit 1 Reactor Zone ventilation system to this condition?

a. Fan C001A trips and supply suction fan valve F024A closes. Fan C004A trips and discharge valves F043A/B close. Fan C007A trips and discharge valves F044A/B close.
b. Fan C001A trips and supply suction fan valve F024A closes. Fan C004A trips and discharge valves F043A/B close. Accessible to inaccessible area bypass valve F027 receives a close signal.
✓c. Fan C004A trips and discharge valves F043A/B close. Accessible to inaccessible area bypass valve F027 opens. Supply suction fan valve F024A throttles partially closed.
d. Fan C004A trips and discharge valves F043A/B close. Fan C007A trips and discharge valves F044A/B close. Accessible to inaccessible area bypass valve F027 opens to cross connect to SBGT.

New question
SI-LP-01303-00, p. 29, 30

32. There is a General Emergency on Unit 2; wind direction is currently from 430°. Values over the last 15 minutes have been in a band from 430° to 470°. Which one of the following sectors would be in the direct path of the release for there wind conditions? (References included)

a. E-10
✓b. G-10
c. J-10
d. L-10

Provide copy of Attachment 4 to 73EP-EIP-054-0S as a reference.

SRO only
New question
LR-LP-20017-02, p. 27
73EP-EIP-054-0S, Attachment 4, p.1

33. Which one of the following describes the basis for the offsite radioactivity release rate parameter used as an entry condition to Unit 2 EOP flowchart RR?

a. It indicates a primary system break which cannot be isolated.
b. It corresponds to an entry into a Site Area Emergency in the Emergency Plan.
c. It represents a release rate that is an immediate threat to the continued health and safety of the public.
✓d. It represents a release rate that is higher than expected during normal plant operations, but does not pose an immediate threat to the public.

SRO only
New question
LR-LP-20325-05, pp. 26,29

34. Given the following Unit 1 plant conditions:

A plant transient has occurred resulting in high RPV pressure.
The SRVs are actuating in the Low Low Set (LLS) mode to relieve pressure.

Which one of the following describes how the LLS signal will reset?

a. The LLS logic automatically resets only when RPV pressure is less than 1080 psig <u>and</u> SRV tailpipe pressure switches are less than 85 psig.
b. The LLS logic automatically resets when RPV pressure is less than 1080 psig <u>or</u> SRV tailpipe pressure switches are less than 85 psig.
c. The LLS logic can be manually reset only if RPV pressure is less than 1080 psig <u>and</u> SRV tailpipe pressure is less than 85 psig.
✓d. The LLS logic can be manually reset if RPV pressure is less than 1080 psig <u>or</u> SRV tailpipe pressure is less than 85 psig.

New question
SI-LP-01401-00, pp. 12, 45

35. Unit 2 was operating at MOP when the main turbine inadvertantly tripped. The following conditions were noted on 2H11-P603 two minutes after the scram:

All 4 scram Group A lights are *illuminated*
All 4 scram Group B lights are *extinguished*
Reactor pressure peaked at 1190 psig and is now 920 psig
RWM shows all control rods are inserted

Which one of the following states the reason why control rods were inserted?

✓a. ARI actuated.
b. Backup scram valves actuated.
c. High reactor pressure scram signal.
d. Main Turbine trip > 30% scram signal.

'97 NRC exam, Q# 65 (slightly modified)
SI-LP-00101-00, p. 16

36. Unit 2 was operating at end of cycle when it scrammed due to a Main Turbine trip. The SS entered EOP flowcharts RC and PC-1 in response to the transient. The following plant conditions exist:

HPCI system:	Out of service
Reactor pressure:	920 psig
Reactor water level:	-25 inches, decreasing slowly
Torus water temperature:	115°F, increasing slowly
Torus water level:	125 inches, decreasing slowly
MSIV status:	Open

As a result of the transient, a large leak has occurred from a bottom weld on the torus. The SS has concluded that the Heat Capacity Temperature Limit will be exceeded in the near future. Which one of the following actions is required by the EOPs for these conditions?

a. Maintain RPV pressure between 850 psig and 1050 psig using SRVs.
b. Begin a controlled pressure reduction using bypass valves; cooldown rates may not be exceeded.
c. Begin a pressure reduction using sustained opening of SRVs; cooldown rates may not be exceeded.
✓d. Rapidly depressurize using bypass valves, irrespective of cooldown limits.

SRO only
New question
LR-LP-20308-07, p. 64
EOP Flowchart RC, C-2

37. The Unit 2 reactor has received a spurious scram signal. During recovery actions the crew identifies that reactor water level is 105 inches. Based on this condition, which one of the following actions should the operators <u>immediately</u> perform?

✓a. Close the MSIVs.
b. Isolate HPCI and RCIC.
c. Trip any operating CRD pump.
d. Reduce reactor water level using RWCU.

'97 NRC exam, Q# 69 (slightly modified)
LR-LP-20301-03, p.14

38. Unit 2 is conducting a plant startup per 34GO-OPS-001-2S, "*Plant Startup,*" and surveillance 34SV-E41-005-2S, "*HPCI Pump Operability 165 PSIG Test,*" is in progress. The following plant conditions exist:

Suppression pool water temperature is 98°F and rising.
Both loops of suppression pool cooling are in service.

The SS is implementing the actions of 34AB-T23-003-2S, "*Torus Temperature Above 95°F.*" In accordance with Unit 2 Technical Specifications, which one of the following actions is required once suppression pool temperature exceeds 105°F?

a. Place the reactor mode switch in SHUTDOWN.
✓b. Suspend all testing that adds heat to the suppression pool.
c. Enter Tech Spec 3.0.3 and commence a controlled shutdown.
d. Depressurize the reactor vessel to less than 200 psig within 12 hours.

SRO only
New question
Unit 2 Tech. Spec. 3.6.2.1
34AB-T23-003-2S, p. 2
LT-LP-20201-05, p. 22

39. During an accident on Unit 2, suppression pool water level has reached 200 inches. Reactor pressure is 300 psig and decreasing. Which one of the following containment components will not properly function at this point?

a. Suppression chamber spray nozzles.
✓b. SRV tail pipes and/or supports.
c. Suppression chamber-drywell vacuum breakers.
d. Normal control room suppression pool level instrumentation.

New question
EOP graph 6
LR-LP-20310-05, pp. 25, 30, 32

40. Unit 2 was operating at 68% power when an unisolable main steam line leak occurred. The following plant conditions exist:

Steam tunnel area temperature:	145°F
130' Northwest area radiation:	1100 mr/hr
130' Southwest area radiation:	820 mr/hr

Based on current plant conditions, which one of the following actions should the operators perform?

✓a. The reactor must be scrammed only.
b. The reactor must be shutdown per 34GO-OPS-013-2S, Normal Plant Shutdown.
c. The reactor must be shutdown per 34GO-OPS-014-2S, Fast Reactor Shutdown.
d. The reactor must be scrammed and Emergency Depressurization initiated per CP-1.

SRO only
'93 NRC exam, Q# 83 (reordered)
LR-ST-20325-02, p. 10

41. From the following list of Safeguard Equipment Cooling coolers, which set will not generate a SEC AUTO INITIATION SIGNAL PRESENT annunciator on panel P650 after they are automatically started?

   a. HPCI room coolers.
   b. RCIC room coolers.
   ✓c. CRD diagonal coolers.
   d. Core Spray and RHR diagonal coolers.

   New question
   SI-LP-01303-00, pp.34-35

42. An ATWS has occurred on Unit 2. Reactor water level is being controlled using the "2A" Reactor Feedwater Pump and drywell pressure is steady at 2.4 psig. The SS directs the PO to prevent the HPCI system from injecting to the RPV. Which one of the following actions is the correct method for accomplishing this task?

a. Manually trip HPCI, then close the HPCI Steam Supply Valve (2E41-F001).
b. Close the HPCI Steam Supply Valve (2E41-F001), then place the HPCI Auxiliary Oil Pump in "Pull-to-Lock".
c. Place the HPCI Auxiliary Oil Pump in "Pull-to-Lock," then press the HPCI manual trip pushbutton until the HPCI turbine has stopped.
✓d. Press the HPCI manual trip pushbutton until the HPCI turbine has stopped, then place the HPCI Auxiliary Oil Pump in "Pull-to-Lock".

Bank question (reworded and reordered)
34SO-E41-001-2S, p. 26
SI-LP-00501-01, p. 7

43. Which one of the following describes the HPCI system suction valve response if the Condensate Storage Tank (CST) level decreases to 30 inches?

a. The Suppression Pool suction valves (E41-F041 & F042) open when the CST suction valve (E41-F004) is fully closed.
b. The Suppression Pool suction valves (E41-F041 & F042) open when the CST suction valve (E41-F004) indicates not fully open.
✓c. The CST suction valve (E41-F004) closes when both Suppression Pool suction valves (E41-F041 & F042) are fully open.
d. The CST suction valve (E41-F004) closes when either Suppression Pool suction valve (E41-F041 or F042) indicates not fully closed.

Bank question (reworded slightly)
SI-LP-00501-01, pp. 10, 14, 22

44. Unit 1 is initially operating at 85% power when Main Turbine Bypass valve "A" suddenly fails full open. With no other failure conditions, which one of the following describes how reactor power responds to this event during the first minute?

a. Reactor power will increase slightly and then decrease to about 85% power.
b. Reactor power will increase and stabilize at about 93% power.
✓c. Reactor power will decrease slightly and then increase to about 85% power.
d. Reactor power will decrease and stabilize at about 77% power.

New question
SI-LP-01901-00, p. 13

45. Unit 1 is initially operating at 90% power when the following plant indications and conditions occur:

> HEATER TROUBLE annunciator is lit
>
> The final feedwater temperature entering the RPV prior to the event was 380°F.
>
> The final feedwater temperature entering the RPV after to the event is 350°F.
>
> There are no known fuel leakers on Unit 1.

Which one of the following is the minimum power reduction, if any, required for these plant conditions?

a. Maintain power at 90%.
✓b. Reduce power 10% using Recirc pumps.
c. Reduce power 20% using Recirc pumps.
d. Reduce power 25% using Recirc pumps.

New question
34AB-C21-001-2S, p. 2

46. Which one of the following statements describes one reason why the Mode Switch is taken from the SHUTDOWN position to the REFUEL position during an ATWS condition?

   a. Allows the scram solenoids to be de-energized without causing MSIV closure.
   b. Allows bypassing the RWM so the operator may drive rods using Emergency In.
   ✓c. Allows control rod selection for position monitoring during individual rod movement or scrams.
   d. Allows the scram to be reset and the scram discharge volume vent and drain valves to be opened.

Bank question (modified distractors)
LR-LP-20314-02, pp. 19-20

47. Unit 2 is operating at MOP with the following conditions:

"A" Station Service Air Compressor (SSAC):	NORMAL
"B" SSAC:	STOP
"C" SSAC:	Running
Service Air pressure:	Normal

A fault on 4160V Bus "2E" results in loss of power to the bus.

Which one of the following describes the expected response of system air pressure and the Station Service Air Compressors? (Assume no operator action.)

a. System pressure will decrease until "A" SSAC automatically starts.
b. System pressure will decrease until "B" SSAC automatically starts.
✓c. System pressure will be maintained, however there is no automatic backup available.
d. System pressure will be maintained, and an automatic backup compressor is available.

New question
LT-03501-03, pp. 17-18

48. Which one of the following describes the condition and an adverse effect associated with operating with a low reactor water level while at power?

a. Increased levels of moisture in the steam can erode turbine blades.
b. Increased levels of moisture in the steam can cause main steam line water hammer.
c. Steam being entrained in the water can cause localized power peaks.
✓d. Steam being entrained in the water can erode recirculation pump impellers.

New question
LT-LP-00202-03, p. 9

49. Unit 1 is operating at 60% power when an event occurs resulting in the following plant conditions:

"A" Reactor Feedwater pump flow:	16%
"B" Reactor Feedwater pump flow:	28%
Reactor water level:	30 inches

Which one of the following describes the response of the Reactor Recirculation Pumps to this situation?

✓a. The Recirc Pumps do not run back.
b. The Recirc Pumps run back to 22%, and reset automatically when "A" feedwater flow increases above 20%.
c. The Recirc Pumps run back to 33%, and reset automatically when "A" feedwater flow increases above 20%.
d. The Recirc Pumps run back to 33%, but must be manually reset when "A" feedwater flow increases above 20% and RPV level increases above 32 inches.

'93 NRC exam, Q# 5 (modified)
SI-LP-00401-00, p. 33

50. Which one of the following is the suppression pool level at which the Unit 2 downcomers become uncovered?

- ✓a. 98 inches
- b. 102 inches
- c. 110 inches
- d. 146 inches

New question
LR-LP-20310-05, pp. 17, 20

51. Unit 1 EOP flowchart PC-1 has the operators perform the following action if suppression pool water level can not be maintained above 115 inches:

> Trip and prevent operation of HPCI irrespective of adequate core cooling

Which one of the following HPCI system responses will this action prevent?

a. Unstable HPCI operation.
b. HPCI exhaust check valve chatter.
c. Loss of back pressure on the exhaust line.
✓d. Overpressurization of the primary containment.

'97 NRC exam, Q# 61
LR-LP-20310-05, p. 23

52. Unit 1 is operating at 25% power when RPS bus "A" is deenergized. Which one of the following statements describes the effect on plant operation due to this event?

✓a. The inboard Reactor Building ventilation dampers will close.
b. The outboard Refueling Floor ventilation dampers will close.
c. A reactor trip will occur if MSIV testing is in progress with one valve closed in MSL "C" when power is lost.
d. A reactor trip will occur if Turbine Stop Valve testing is in progress with one TSV closed when power is lost.

New question
SI-LP-01001-01, p. 27, 34-36

53. A partial loss of AC power has occurred resulting in loss of one RPS bus. The SS desires to re-energize the RPS bus from its alternate power supply. Which one of the following statements correctly describes how alternate power is supplied to the RPS buses?

a. RPS Bus "B" may receive its alternate power supply from Instrument Bus "A" or "B" depending on the position of the RPS Power Source Select Switch on P610.
b. RPS Bus "A" or RPS Bus "B" may receive its alternate power supply from Vital AC after repositioning the throwover switch in the RPS MG Set room.
✓c. RPS Bus "B" may receive its alternate power supply from Essential Cabinet "A" or "B" depending on the position of the throwover switch in the RPS MG Set room.
d. RPS Bus "A" receives its alternate power supply from Essential Cabinet "A" and RPS Bus "B" receives its alternate power supply from Essential Cabinet "B".

Bank question (modified)
SI-LP-01001-01, pp. 28, 45

54. Unit 2 is operating at MOP with all Plant Service Water pumps running. Annunciator TURB BLG PSW FLOW HIGH actuates and the operators note that both Division 1 and Division 2 PSW pressures are reading 45 psig. Based on these conditions, which one of the following is the proper operator response?

✓a. Manually scram the reactor and close valves 2P41-F316A, B, C, and D.
b. Throttle closed 2P41-F316A and B until division pressures are > 80 psig.
c. Reduce reactor power as required to maintain equipment temperatures within limits.
d. Reduce reactor power as required to maintain equipment temperatures within limits and close 2P41-316A, B, C, and D.

'97 NRC exam, Q# 74 (slightly modified)
34AB-P41-001-2S, p. 3
LT-LP-20201-05, pp. 2, 16

55. A startup of Unit 2 is in progress with no equipment out of service. Reactor power is 40% and the speed of both recirc pumps has just been raised to 30%. A trip of Recirc Pump "2A" occurs and the operators respond to the transient per the guidance of 34AB-B31-001-2S, "*Reactor Recirculation Pump(s) Trip, or Recirc Loops Flow Mismatch*" to stabilize the plant. Which one of the following describes how an accurate reading of total core flow is determined under these conditions?

a. The Total Core Flow indication must be reduced by the "2A" Jet Pump flow to obtain an accurate reading.
✓b. Total core flow must be manually calculated by adding "2A" and "2B" Jet Pump flows to obtain an accurate reading.
c. Total core flow must be manually calculated by subtracting "2A" Jet Pump flow from the "2B" Jet Pump flow to obtain an accurate reading.
d. The summing circuitry for the Total Core Flow indication automatically accounts for the idle "2A" recirc loop and provides an accurate reading.

New question
34AB-B31-001-2S, p. 2, Note
SI-LP-00401-01, p. 19

56. Unit 1 is operating at 55% power and rod scram insertion time testing is in progress. While withdrawing control rod 42-18 from position 28 to 48, rod movement suddenly stops and the following plant conditions are observed:

CHG WATER PRESS HIGH annunciator is lit
CRD HYD HIGH TEMP annunciator is lit

CRD system flow:	10 gpm
Cooling water flow:	11 gpm
Charging water pressure:	1570 psig
Drive water dP:	15 psid
Cooling water dP:	0 psid

Which one of the following is the cause of the above plant indications?

a. The CRD pump tripped.
b. The flow stabilizing valves failed closed.
✓c. The CRD flow control valve failed closed.
d. The cooling water pressure control valve failed closed.

Lesson plan question (reworded and re-arranged)
LT-LP-00101-04, pp. 20-21

57. Given a loss of 125/250 VDC Switchgear "B", which one of the following systems would be unavailable?

   a. RCIC
   ✓b. HPCI
   c. LPCI mode of RHR
   d. 2A EDG

   New question
   LT-LP-02704-03, p. 38

58. Unit 2 is operating at MOP when a plant air system break occurs. Upon investigation, the operators determine the following information:

Service Air Pressure	Non-Essential Air Pressure	Interruptible Essential Air Pressure	Non-Interrupt Essential Air Pressure
0 psig	0 psig	50 psig	105 psig

Based on these indications, which one of the following describes the most likely location of the rupture? (References included)

a. Service air header line
✓b. Non-essential air header line
c. Interruptible essential air header line
d. Non-interruptible essential air header line

Provide a simplified one line diagram of IA system as a reference.

'97 NRC exam, Q# 71
LT-LP-03501-03, p. 51

59. Unit 2 is operating at MOP. All condensate and condensate booster pumps are running and the "A" SJAE is in service. Suddenly, the "2C" Condensate Pump trips. When conditions stabilize, the following conditions exist: (Assume no operator action.)

COND PUMPS DISCH PRESS LOW annunciator is lit
Condensate discharge pressure indicator reads 95 psig
Main Condenser vacuum is slowly decreasing

Which one of the following decribes the cause of the vacuum decrease?

a. Closure of Main Steam Supply to SJAE, 2N11-F001.
b. Closure of Condenser Inner Suction Valve, 2N22-F004A.
c. Closure of Condenser Outer Suction Valve, 2N22-F005A.
✓d. Closure of First Stage Steam Supply Valve, 2N11-F008A.

'97 NRC exam, Q# 68 (slightly modified)
SI-LP-02501-00, p. 8

60. Unit 2 is in Mode 3 with the "2A" RHR pump in shutdown cooling when valve 2E11-F008 spuriously closes and will not re-open. Which one of the following is the appropriate operator response to this event?

a. Place the "2C" RHR pump in the Shutdown Cooling Mode of operation.
b. Place the "2B" loop of RHR in the Shutdown Cooling Mode of operation.
✓c. Increase reactor water level greater than 53 inches to promote natural circulation.
d. Throttle open the 2E11-F017A, RHR Outboard Injection Valve, to increase cooling.

'97 NRC exam, Q# 72 (slightly modified)
LT-LP-20201-05, p. 14

61. Unit 2 has been in Mode 4 for 8 weeks for "2A" Core Spray Pump replacement. Maintenance has been completed and it is decided to place the Core Spray system Loop "A" in standby condition. A valve line up for this subsystem has been completed. Based on these conditions, which one of the following statements list the minimum additional administrative requirements, if any, that must be met for placing the loop in standby?

✓a. Both an instrument valve line up and an electrical line up are required to be done prior to placing the loop in standby.
b. An instrument valve line up is not required unless the SOS requires it to be done; an electrical line up is required to be done prior to placing the loop in standby.
c. An instrument valve line up is required to be done prior to placing the loop in standby; an electrical line up is not required unless the SOS requires it to be done.
d. Neither an instrument valve line up nor an electrical line up are required to be done unless the SOS requires they be done prior to placing the loop in service.

SRO only
Bank question (reworded)
34GO-OPS-003-2S, p. 3, sect. 7.0
SI-LP-0801-00

62. Unit 2 has just completed a refueling outage. While placing the Mode Switch in START & HOT STBY per 34GO-OPS-001-2S, "*Plant Startup,*" the mode switch is inadvertantly positioned to RUN. Which one of the following is the expected plant response?

a. The reactor will not scram and the MSIVs will remain open.
b. The MSIVs will close but the reactor will not scram.
c. The reactor will scram but the MSIVs will remain open.
✓d. The MSIVs will close and the reactor will scram.

New question
34GO-OPS-001-2s
SI-LP-01401-00, p. 18, 23

63. Unit 2 is operating at 50% power with Feedwater Level Control in 3 element. The Reactor Level Select Switch is in "A" Level position. The following indications are observed:

RFPT speed:	stable
Total feedwater flow:	stable
Indicated level on R606A:	25 inches, decreasing
Indicated level on R606B:	37 inches, stable
Indicated level on R606C:	24 inches, decreasing

Which one of the following plant responses will occur based on the above indications?

a. The reactor will scram due to Turbine Trip.
✓b. The reactor will scram due to low reactor water level.
c. The reactor will not scram due to the feedpump TMR shifting to Speed Setter Mode of operation.
d. The reactor will not scram due to the RWLC system shifting to "B" Level as median reactor water level control.

Bank question (modified)
34AB-B21-002-2S
SI-LP-04404-00, p. 32, 35

64. Unit 2 is operating at 23% power. Drywell venting is in progress via valves 2T48-F319 and 2T48-F320. Drywell pressure is 0.6 psig. During this time, the Unit 1 Refuel Floor Vent Exhaust Radiation Monitors, 1D11-D611A thru D reach their trip setpoints. Which one of the following describes the effect of the Unit 1 radiation monitors on the Unit 2 drywell venting?

a. Venting would continue with Unit 2 Standby Gas Treatment System taking suction on the Unit 2 drywell only.
✓b. Venting would continue with Unit 2 Standby Gas Treatment System taking suction on the Unit 2 drywell, the Unit 2 reactor building and the refuel floor.
c. Venting would stop due to Unit 2 vent and purge valves (F319 and F320) closing.
d. Venting would stop due to Unit 2 filter train suction dampers realigning to take suction from the Unit 2 reactor building and the refuel floor.

'93 NRC exam, Q# 17
SI-LP-01301-00, p. 35
SI-LP-01302-00, p. 16

65. A fire alarm is received in the control room on the CXL for the Cable Spreading Room. The PEO investigates the alarm and reports that black smoke is coming out from around the door to the Cable Spreading Room. He also reports that the red light next to the CO_2 "START" pushbutton is extinguished. Which one of the following describes the actions required, if any, to discharge CO_2 into the cable spreading room?

a. No actions required, the extinguished red light indicates that CO_2 has been automatically released into the room.
b. The extinguished red light indicates that automatic discharge of CO_2 has failed and the Fire Brigade Leader must depress the "START" pushbutton for CO_2 to be released into the room.
✓c. When the Fire Brigade Leader operates the manual release lever on the Master Pilot Valve, CO_2 will be discharged into the room.
d. When the Fire Brigade Leader presses the "START" pushbutton and the red light illuminates, CO_2 will be discharged into the room.

Bank question (modified)
LT-LP-03601-03, pp. 53-56

66. A large break LOCA has occurred on Unit 1. Which one of the following potential consequences could occur if the Suppression Chamber-Drywell Vacuum Breakers failed open during this event?

a. When drywell sprays are initiated, drywell pressure will decrease such that the external design pressure of the drywell will be exceeded.
b. When the drywell blows down to the torus, the radioactive gases from the suppression pool will be released directly to the Reactor Building atmosphere
✓c. When the drywell blows down to the torus, the steam will pass straight through to the torus air space resulting in primary containment pressure exceeding internal design pressure.
d. When torus sprays are initiated, the non-condensible gases released from the suppression pool will be vented directly back to the drywell resulting in a rapid increase in drywell pressure.

Bank question (modified slightly)
SI-LP-01301-00, pp. 11, 40

67. Given the following plant conditions:

- A LOCA has occurred
- "2B" RHR pump is the only high volume source of water
- The room coolers for "2B" RHR pump will not run
- "2B" RHR pump and motor temperatures are increasing
- A maintenance worker needs to enter the diagonal to set up temporary cooling
- The task will take no longer than 20 minutes, radiation levels are 30 R/hr.

Which one of the following describes the approval needed, if any, for the worker to enter this area under these radiation conditions?

a. Prior approval is not required because the dose would be within the predefined Plant Hatch emergency response personnel exposure limits.
b. Prior approval is not required because the dose would be within NRC limits, but a 10 CFR 50.72 report would be required.
c. The Senior Vice President of Nuclear Operations must give approval prior to performing the task.
✓d. The Emergency Director must give approval prior to performing the task.

SRO only
New question
73EP-EIP-017-0S, p. 4
LT-LP-30008-02, p. 11

68. Given the following exposure history for an 21 year old male radiation worker:

Lifetime exposure:	14500 mrem (Form 4 on file)
Annual exposure:	4300 mrem
Quarterly exposure:	600 mrem

Which one of the following statements describes the maximum additional whole body dose the individual is allowed in the current calendar quarter per 10 CFR 20 exposure limits?

a. 400 mrem
b. 500 mrem
c. 650 mrem
✓d. 700 mrem

Bank question (slightly modified, answer changed due to 10CFR20 changes)
10 CFR 20.1201
LT-LP-30008-02, p. 7
60AC-HPX-001-0S, p. 5

69. Unit 2 reactor shutdown is in progress and primary containment de-inerting has been authorized. Which one of the following is the basis for <u>not</u> allowing <u>all</u> four 18 inch containment air vent valves (2T48-F318, F319, F320, and F326) to be open simultaneously during the performance of this evolution?

a. To prevent the high flow rate from damaging the non-hardened ventilation ducts.
b. To prevent creating a high dP between the primary containment and the Reactor Building.
✓c. To prevent the possibility of any drywell steam from entering the torus air space at power.
d. To prevent release of the drywell atmosphere through an unmonitored ventilation flow path.

New question
34SO-T48-002-2S, 1st Caution, p. 16
SI-LP-01301-00, p. 32

70. Given the following conditions for Unit 2:

Post Treatment Radiation Monitor K615A is out of service for maintenance
Post Treatment Radiation Monitor K615B receives a valid Hi-Hi signal

POST TREATMENT OFFGAS RADIATION HI-HI annunciator is lit
POST TREATMENT OFFGAS RADIATION HI annunciator is lit

Which one of the following describes the complete expected response of the Offgas system?

a. No automatic actions will occur.
✓b. Carbon bed bypass valve (F043) closes and carbon bed inlet valve (F042) opens.
c. Offgas Stack Isolation (2N62-F057), Offgas Cooler Condenser/Moisture Separator Valves (N62-F030A and B both units), and Offgas Holdup Line Drain, (2N62-F085) will close.
d. Carbon bed bypass valve (F043) closes and carbon bed inlet valve (F042) opens. Offgas Stack Isolation (2N62-F057), Offgas Cooler Condenser/Moisture Separator Valves (N62-F030A and B both units), and Offgas Holdup Line Drain, (2N62-F085) will close.

New question
LT-LP-10007-04, p. 19

71. The Unit 1 Floor Drain Sample Tank is being released through valves 1G11-F428 and 1G11-F430 to the Discharge Canal per 34SO-G11-036-1S. The Radwaste Canal Discharge Line Isolation Valves, 1G11-F184 and 1G11-F185, are open to support the release. Which one of the following describes the complete response of the Radwaste system if, during the release, the Liquid Radwaste Effluent Radiation Monitor receives a high radiation trip signal?

✓a. Both Radwaste Canal Discharge Line Isolation Valves (1G11-F184 and 1G11-F185) close only.
b. The FDST pump trips and the outboard Radwaste Canal Discharge Line Isolation Valve (1G11-F185) closes only.
c. The FDST pump trips, the tank discharge isolation valve (1G11-F428) closes, and the outboard Radwaste Canal Discharge Line Isolation Valve (1G11-F185) closes.
d. The FDST pump trips, the tank discharge isolation valve (1G11-F428) closes, and both Radwaste Canal Discharge Line Isolation Valves (1G11-F184 and 1G11-F185) close.

New question
34SO-G11-036-1S
LT-LP-02901-02, pp. 14-20, 32-36
LT-LP-10007-04, p. 26

72. Unit 2 is operating at MOP. The control room receives the following alarm:

RBCCW SURGE TK LEVEL LOW OR EXCESS LEAKAGE

The Unit 2 inside rounds PEO checks the RBCCW system and notes the following equipment status:

RBCCW surge tank counter reads:	0000
RBCCW surge tank timer reads:	08:15
RBCCW surge tank level reads:	45"

Which one of the following statements correctly represents the current condition of the RBCCW system and appropriate PEO response?

a. An automatic make-up is in progress and the local Fill Cycle Timer reset pushbutton needs to be depressed to allow future automatic make-up to occur.
b. A fill occurred 8 hours and 15 minutes ago. If surge tank level is not restored above the make-up valve opening setpoint within the next 4 hours and 45 minutes, then a system leak above 134 gallons is in progress.
c. A fill occurred 4 hours and 45 minutes ago. If surge tank level is not restored above the make-up valve opening setpoint within the next 8 hours and 15 minutes, then a system leak above 134 gallons is in progress.
✓d. A system leak above 134 gallons is confirmed and manual make-up to the tank needs to be initiated.

Bank question (modified slightly)
34AR-650-248-2S
SI-LP-00901-00, pp. 13-14

73. Unit 1 RCIC is injecting to control reactor vessel level. A large oil leak develops on the in-service RCIC oil filter which results in decreasing oil pressure. Which one of the following describes the response of the RCIC system as oil pressure decreases?

a. The governor valve will close and turbine speed will decrease to zero RPM.

✓b. The governor valve will open and turbine speed will increase possibly resulting in a turbine trip.

c. The auxiliary oil pump will start, the trip valve will close and turbine speed will decrease to zero RPM.

d. The turbine will trip but the steam stop valve will fail open on low oil pressure.

'93 NRC exam, Q# 10 (changed units)
SI-LP-03901-00, p. 8, 17

74. Unit 2 is starting up. A plant heatup/pressurization is in progress. Reactor water level control is in "dP" mode, with the "2A" Reactor Feed pump in service. Which one of the following describes the effect of lining up RWCU to blowdown to the Main Condenser while steaming in this condition?

a. It will overheat the Regenerative Heat Exchanger.
✓b. It will improve stability of the Feedwater Control system.
c. It will complicate level control by causing a level increase.
d. It will flood out the Main Condenser Hotwell causing a loss of vacuum.

Bank question (reworded slightly)
34SO-N21-007-2S, p. 37, Note
SI-LP-00201-00, pp. 28-29, 36

75. Unit 2 has experienced a reactor scram from MOP. The following conditions exist:

Highest drywell temperatures:	215 °F (2T47-N001A & N001K)
Drywell pressure:	1.3 psig
Reactor pressure:	920 psig

The following RPV water level instruments read as indicated:

Floodup Range:	+13 inches
Narrow Range:	+ 6 inches
Wide Range:	+ 4 inches
Fuel Zone:	- 80 inches

Based on the above conditions, which one of the following reactor water level indicators would be considered unreliable for level trend information per EOP guidance? (References included)

✓a. Floodup Range
b. Narrow Range
c. Wide Range
d. Fuel Zone

Provide copy of Attachment 1 (Caution 1) to 34AB-B21-002-2S as a reference.

Bank question (reworded, changed floodup range initial condition)
34AB-B21-002-2S, Att. 1, pp. 2-3
LR-LP-20305-04, pp. 7-14

76. Which one of the following specifies how adequate NPSH is ensured for the Recirc Pumps when the Unit 1 reactor is operating at 5% power?

Physical placement of the Recirc Pumps and:

✓a. the # 1 speed limiter.
b. the # 2 speed limiter.
c. subcooling of the downcomer water due to feedwater.
d. subcooling of the downcomer water due to carryunder.

Bank question (reordered and reworded slightly)
SI-LP-00401-01, p. 9

77. Which one of the following is the reason that continued plant operation with an inoperable (or failed) jet pump is restricted?

   a. Invalid APRM flow biased scram setpoints due to the change in flow through the failed jet pump.
   ✓b. Increased blowdown area during a LOCA.
   c. Unbalanced neutron flux across the core due to flow variations.
   d. Physical core and cladding damage from a loose piece of the damaged jet pump.

'93 NRC exam, Q# 23
SI-LP-00401-01, p. 15

78. While moving an irradiated fuel bundle from the East Fuel Prep Machine to its storage location in the Unit 2 fuel storage rack, the bundle is dropped. The bundle hits on top of the rack and then falls to the bottom of the pool. The bridge operator observes bubbles rise out of the water and refuel floor area radiation monitors begin alarming. Which one of the following actions should be immediately performed by the Refueling SRO per 34AB-J11-001-2S, *"Irradiated Fuel Damage During Handling"*?

a. Evacuate all personnel from the reactor building.
b. Start the Standby Gas Treatment system.
c. Isolate the Secondary Containment.
✓d. Cease all refuel floor operations.

SRO only
Bank question (modified)
34AB-J11-001-2S, p. 1
LT-LP-04502-03, p. 36

79. Which one of the following describes the purpose of the Drywell Spray Initiation Limit curve (EOP Graph 8)?

a. It ensures evaporative cooling occurs to maximize spray effectiveness.

✓b. It ensures that initiation of drywell sprays will not result in containment failure.

c. It ensures initiation of drywell sprays for dilution of drywell H_2 and O_2 concentration.

d. It ensures the suppression chamber-drywell vacuum breakers will properly function.

New question
LT-LP-20310-05, p. 48

80. Unit 2 has scrammed with the following plant conditions:

Reactor pressure:	920 psig, steady
Reactor water level:	0 inches, increasing slowly
Drywell pressure:	4 psig
Drywell temperature:	220°F, increasing slowly
Torus level:	149 inches
Torus temperature:	93°F, increasing slowly

The unit subseqently experiences a sustained loss of 125/250 VDC Bus "A". Due to level being controlled by HPCI/RCIC, the crew manually secures the "2A" and "2C" RHR pumps and places the "B" loop of RHR in torus cooling. If an additional loop of torus cooling is later needed, which one of the following actions is required to start the "2A" RHR Pump?

a. Place the "2A" RHR pump control switch to the START position.
✓b. Depress the LOCA reset pushbutton for RHR pump "2A".
c. Place the "2A" RHR pump control switch to the STOP, then to the START position.
d. Place the LOCA override switch to MANUAL, then place the "2A" RHR pump control switch to the START position.

New question
SI-LP-00701, p. 23

81. Unit 2 Loop "A" of RHR is in full flow test with 2E11-F028A OPEN and 2E11-F024A THROTTLED. A large break LOCA and a reactor scram occurs while the test is in progress. With reactor water level at -90 inches (decreasing) and a torus pressure of 9 psig (increasing), the SS directs the operator to initiate torus spray.

If the operator takes the Containment Spray Valve Control Switch to MANUAL and the torus spray valve (2E11-F027A) to open, which one of the following describes the expected system response? (Assume no other manual operator actions were taken during the transient and all ECCS equipment responds automatically, as designed.)

✓a. The torus spray valve will open immediately and design flow will spray the torus when 2E11-F028A strokes open.
b. The torus spray valve will open immediately but no spray flow will occur until 2E11-F028A is manually opened.
c. The torus spray valve will remain closed until the 2/3 core height interlock is manually overridden, then design flow will spray the torus when 2E11-F028A strokes open.
d. The torus spray valve will remain closed until the 2/3 core height interlock is manually overriden but no spray flow will occur until 2E11-F028A is manually opened.

New question
EOP Flowchart PC-1
SI-LP-00701-00, pp. 24-25, 32, 48

82. Unit 1 is operating at 75% power with the "D" APRM bypassed. Which one of the following describes the effect on RBM system if the "B" APRM fails low?

a. Both RBM channels are bypassed.
b. RBM Channel A is selected to APRM "A" and Channel B is bypassed.
c. RBM Channel A is bypassed and Channel B is selected to APRM "C".
✓d. RBM Channel A is selected to APRM "A" and Channel B is selected to APRM "C".

New question
SI-LP-01203-00, pp. 21-22

83. A loss of feedwater and large break LOCA have occurred on Unit 2. The following plant indications currently exist:

Reactor level:	-175 inches and steady
Reactor pressure:	53 psig
SRV status:	7 ADS valves OPEN
RHR pump status:	2A injecting at 11,500 gpm
CS pump status:	2A tagged out 2B indicates 0 gpm, discharge pressure is oscillating between 50 psig and 320 psig
Torus level:	142 inches
Torus water temperature:	165 °F

Based on the above conditions, which one of the following actions should the operator perform? (References included)

a. Reduce 2A RHR flow to within the NPSH limit.
✓b. Align the suction of the 2B CS pump to the CST.
c. Maintain the current status until RPV level is above TAF.
d. Throttle closed 2E21-F005B to increase CS discharge pressure.

Provide copies of Graphs 12A & 12B, "RHR Pump NPSH Limit" (Suppression Pool Water Level At or Above and Below 146) as references.

SRO only
'97 NRC exam, Q# 60
34AB-E10-002-2S, p. 2
LR-LP-20309-05, pp. 13-14

84. An ATWS has occurred on Unit 2 and reactor power is approximately 19% power. The operator attempts to insert control rod 22-27 with the EMERGENCY IN switch but the rod fails to move. The operator then notes the following plant conditions:

Drive water D/P:	260 psig
Rx Mode Switch:	REFUEL
CRD flow:	> 100 gpm
CRD FCVs:	CLOSED
CRD Pumps:	"2A" and "2B" running
RWM:	Normal

Which one of the following describes the reason why control rod 22-27 will not move?

✓a. The RWM is enforcing an insert block.
b. There is excessive CRD flow to the HCU accumulators.
c. The drive water D/P is not sufficient to move the control rod.
d. The CRD flow to the HCU accumulators is shut off because the CRD FCVs are closed.

'97 NRC exam, Q# 27 (slightly reworded and rearranged)
SI-LP-05401-00, p. 7

85. Unit 2 is operating at MOP. Which one of the following describes that expected response of the Backup Scram Valves and the Scram Pilot Solenoid Valves to a loss of RPS bus "B"?

a. One backup scram valve energizes and half of the scram pilot solenoid valves deenergize.
b. One backup scram valve energizes and all scram pilot solenoid valves remain energized.
✓c. Both backup scram valves remain deenergized and half the scram pilot solenoid valves deenergize.
d. Both backup scram valves remain deenergized and all the scram pilot solenoid valves remain energized.

Bank question (reordered and reworded slightly)
SI-LP-01001-01, pp. 20-21

86. Unit 2 is operating at MOP with the "A" EHC Pressure Regulator in service when the #4 Turbine Control Valve goes closed. Which one of the following describes the expected plant response to this event. (Assume no operator action.)

a. The reactor scrams on high reactor pressure.
✓b. The turbine bypass valves open to control pressure.
c. The EHC pressure regulator shifts to "B" controlling.
d. The reactor scrams due to a turbine trip signal from the TCV closure.

'97 NRC exam, Q# 1
SI-LP-01901-00, pp. 9-13

87. SRV operability testing is in progress with Unit 2 operating at 30% power. After SRV 2B21-F013A is opened for the test, it remains open even though the control switch was cycled several times and then placed to AUTO. The operating crew implements the actions of 34AB-B21-003-2S, *"Failure of Safety/Relief Valves,"* and the following is the current status of the plant:

SAFETY/BLOWDOWN VALVE LEAKING annunciator is lit
TORUS WATER TEMP HIGH annunciator is lit

Suppression Pool temperature:	105 °F, increasing slowly
Torus cooling:	1 loop of RHR aligned
2B21-F013A control switch:	AUTO
2B21-F013A status lights:	Green NOT LIT Amber LIT Red NOT LIT
2B21-F013A fuses:	Removed
2B21-F013A discharge temperature:	290 °F, increasing slowly

Based on the current plant conditions, which one of the following actions is required?

a. Maximize torus cooling only.
b. Reset the Low Low Set Logic to attempt to close the valve.
c. Commence a fast reactor shutdown per 34GO-OPS-014-2S.
✓d. Scram the reactor per 34AB-C71-001-2S.

Bank question (modified slightly)
34AB-B21-003-2S, p. 3
SI-LP-01401-00, pp. 9-12, 16-18

88. A reactor scram has occurred on Unit 2. The SS directs the operator to verify that all control rods have fully inserted by obtaining a "Control Rod Position" printout from the plant process computer. Which one of the following would be confirmation that the reactor is shutdown?

a. 117 of the control rod positions read "00" and 20 are blank.
✓b. 117 of the control rod positions read "02" and 20 read "00".
c. 121 of the control rod positions read "00", 13 read "02", and 3 read "S".
d. 121 of the control rod positions read "02", 13 read "00", and 3 read "-99".

New question
34AB-C71-001-2S, pp. 1, 18
LR-LP-20301-03, p. 7
LT-LP-40001-02, p. 38

89. An ATWS has occurred on Unit 2. Injection was terminated and prevented for power/level control and eventually re-established. The following conditions now exist:

Reactor level:	-100", increasing slowly
SLC tank level:	10%
Reactor power:	3%

Based on these conditions, which one of the following actions should the crew perform?

✓a. Commence a controlled cooldown of the reactor vessel.
b. Exit the CP-3 flowchart and control level per the RC flowchart.
c. Exit the RCA and CP-3 flowcharts for level, power, and pressure control.
d. Maintain RPV level between -60 and -185 inches until all APRMs are downscale.

SRO only
'97 NRC exam, Q# 62 (modified)
LR-LP-20327

90. An ATWS has occurred on Unit 1. After running back recirculation flow to minimum, reactor power indicates 4% on the APRMs. Which one of the following actions should the SS direct under this condition and the basis for performing it?

a. Trip both recirc pumps to further reduce reactor power.
b. Trip both recirc pumps to remove pump heat from the reactor system heat load.
✓c. Keep both recirc pumps operating to enhance boron mixing during SLC injection.
d. Keep both recirc pumps operating because reactor water level will be too low to establish natural circulation.

SRO only
New question
EOP Flowchart RCA RPV Control ATWS
LR-LP-20328-06, p. 43

91. Unit 1 is at MOP. Both Unit 1 SBGT subsystems have just been returned to operable status following an inspection of the hardened vent rupture disk. The unit's refuel floor equipment hatch is removed.

Unit 2 is in Mode 4. No core alterations are being performed and the refuel floor equipment hatch is in place. However, fuel is being shuffled in the Unit 2 fuel pool. No operations with the potential for draining the reactor vessel are in progress. Both Unit 2 SBGT subsystems are operable.

No refuel floor airspace openings exist on either unit. Zone 1 and Zone 3 are being maintained. No radiation monitors are in alarm. The following plant conditions exist:

Unit 1 drywell pressure:	0.1 psig
Unit 2 drywell pressure:	0.2 psig
Unit 2 reactor temperature:	180°F, vented

A low pressure weather front passes through the area and causes several reactor building roof vents to open. Annunciator RB ROOF VENT/ELEV 280 on P653 is received as a result of the roof vents opening. After 20 seconds, the roof vents close. All equipment functioned normally. Based on the above events and plant conditions, which one of the following Secondary Containment Integrity configurations is now in effect? (References included)

a. Secondary Containment type A is established.
✓b. Secondary Containment type B1 is established.
c. Secondary Containment type B2 is established.
d. Secondary Containment type C is established.

<u>Provide a copy of Section T8.0, "Secondary Containment," (TRM) as a reference.</u>

SRO only
New question
LT-LP-01302-03, pp. 8-10
SI-LP-01302-00, pp. 15-17

92. Unit 2 scrammed due to a loss of condenser vacuum. Neither the Condensate nor Feedwater systems are available and all MSIVs are closed. RCIC is in manual, controlling reactor water level and HPCI is in reactor pressure control. Both loops of torus cooling are in service. The following plant conditions exist:

Reactor water level:	-19 inches, increasing slowly
Reactor pressure:	810 psig, stable
Suppression pool temperature:	140°F, increasing 1°F/2 min.
Torus NE area instrument sump:	High-High-High alarm received
Torus SE area instrument sump:	High-High-High alarm received
Sump pump status:	All operating

The torus sump levels have been in alarm for five hours due to a confirmed RHRSW system leak. Which one of the following actions should the operators perform for these condtions? (References included)

a. Isolate all systems discharging water into the sump except the RCIC system.
b. Isolate RHRSW to the RHR heat exchangers, secure torus cooling and enter the SAGs.
c. Do not isolate any systems discharging water into the sump due to these systems being required to assure adequate core cooling.
✓d. Do not isolate any systems discharging water into the sump due to these systems being required to maintain Primary Containment Integrity.

Provide copy of only the SC/L leg of the SC flowchart as a reference.

SRO only
New question
Hatch LER #99-366-006
EOP Flowchart SC, path SC/L
LR-LP-20305-05

93. A radiological event has occurred on Unit 1 resulting in radiation levels greater than Max Safe Operating Value in the 158' elevation area (north) and the 185' elevation area of the Reactor Building. Which one of the following, in conjunction with the above conditions, would require a reactor scram and Emergency Depressurization to be initiated per EOP flowchart SC?

✓a. An unisolable sample line break occurs at the reactor sample sink.
b. An uncontrolled fire is in progress at Remote Shutdown Panel C82-P002.
c. Severe weather is approaching the site with the wind blowing towards Baxley.
d. 20 rods in the north CRD HCU bank cannot be moved for a reactor power change.

SRO only
New question
EOP Flowchart SC, D-9

94. Both units are operating at MOP when a Hi-Hi alarm is received on Unit 1 Reactor Building exhaust ventilation radiation monitor channels K609A and B. Channels K609C and D indicate normal. Which one of the following describes the response of both units' Secondary Containment systems?

a. No automatic actions occur on either unit.
b. Only Unit 1 SBGT system auto starts. Unit 1 and 2 Reactor Building ventilation trips and only the outboard isolation valves close.
c. Unit 1 and 2 SBGT systems auto start. Unit 1 and 2 Reactor Building ventilation trips and all isolation valves close.
✓d. Unit 1 and 2 SBGT systems auto start. Unit 1 and 2 Reactor Building ventilation trips and only the inboard isolation valves close.

New question
LT-LP-10007-04, p. 28

95. A loss of shutdown cooling has occurred on Unit 2 and Alternate Shutdown Cooling has been established per 34AB-E11-001-2S, "*Loss of Shutdown Cooling,*" using the "A" RHR loop in the LPCI mode. The "B" RHR loop is in Suppression Pool cooling and the "B" SRV is open.

Procedure 34AB-E11-001-2S requires maintaining RPV pressure less than 165 psig above suppression pool pressure. Which one of the following describes the consequence of exceeding this pressure limit?

a. SDC suction isolation valves, 2E11-F008/F009, will auto isolate.
b. The cooldown rate increases, potentially violating the cooldown rate limit.
c. The RPV Pressure-Temperature limit for a non-critical core will be violated.
✓d. RHR pump flow decreases reducing the amount of decay heat that can be removed.

New question (based on BSEP '98 exam question #89)
34SO-E11-010-2S, sect. 5.1.13, p. 4
34AB-E11-001-2S, p. 7, 9
SI-LP-00701-00, p. 21, 29, 33-34

96. Unit 2 reactor startup and heatup is in progress. After verifying SRM/IRM overlap, the SRM detectors were withdrawn per 34GO-OPS-001-2S, *"Plant Startup"*. Which one of the following is correct regarding use of reactor period as an indication to check reactor power response to control rod withdrawal?

a. It is not valid because inputs to the reactor period indicator are automatically bypassed when the IRMs are above range 3.
✓b. It is still valid with the SRMs in the fully withdrawn position because the SRM detectors continue to monitor neutron flux.
c. It is not valid with the SRMs in the fully withdrawn position because the SRM detectors can now only monitoring background radiation.
d. It is still valid because the inputs to the reactor period indicator are automatically transferred to the IRMs when all IRM range switches are above range 3.

Bank question (reworded and reordered)
34GO-OPS-001-2S
SI-LP-01201-00, p. 6

97. Chemistry has just sampled the Unit 2 SLC storage tank and the following conditions were reported to the Shift Supervisor:

Volume	Concentration	Temperature
2000	8%	52°F

Based on these conditions, which one of the following relates the current status and appropriate action for the SLC system? (References included)

a. The system is operable but boron concentration (only) needs to be increased.
b. The system is operable but both boron concentration and temperature need to be increased.
✓c. The system is inoperable and boron concentration (only) needs to be increased.
d. The system is inoperable and both boron concentration and temperature need to be increased.

SRO only
Provide a copy of Unit 2 T.S., Figs. 3.1.7-1 and 3.1.7-2 as a reference.

'97 NRC exam, Q# 14 (modified)
Unit 2 T.S., Figs. 3.1.7-1 and 3.1.7-2
SI-LP-01101-00, pp. 30-31

98. Which one of the following Standby Gas Treatment system components is directly powered by 120/208V Distribution Cabinet 2B Instrument Bus 2A (2R25-S064)?

✓a. SBGT initiation logic supply "A".
b. SBGT initiation logic supply "B".
c. SBGT heat detector and water spray Division I.
d. SBGT heat detector and water spray Division II.

New question
34SO-T46-001-2S, p. 14
SI-LP-03001-00

99. Unit 1 is operating at 75% power when the "B" SRV fails open. The fuses are pulled to the "B" SRV and the following conditions are noted:

SAFETY/BLOWDOWN VALVE LEAKING annunciator lit
SAFETY BLOWDOWN PRESSURE HIGH annunciator green
SPDS indication for the "B" SRV is green
Suppression pool temperature is 111°F

Based on these conditions, which one of the following statements regarding "B" SRV and plant conditions is correct? The "B" SRV is:

a. OPEN and the reactor should be manually scrammed.
✓b. CLOSED but the reactor should be manually scrammed.
c. OPEN and both loops of RHR should be placed in suppression pool cooling only.
d. CLOSED but both loops of RHR should be placed in suppression pool cooling only.

'97 NRC exam, Q# 64
LR-LP-20310-05, p. 37

100. A Unit 2 TIP trace is being run in the Manual mode using TIP Machine C which is in the core at the TOP limit. RPV water level then lowers to 1" and reactor building ventilation exhaust radiation monitors K609A-D begin alarming. Which one of the following describes the TIP system response for this condition?

a. No automatic response will occur because the TIP trace is being run in manual.
b. The shear valve for TIP Machine C will automatically fire to isolate any radioactive release from this pathway.
c. TIP Machine C will automatically withdraw to the in-shield position, then the ball valve must be manually closed.
✓d. TIP Machine C will automatically withdraw to the in-shield position, then the ball valve will automatically close.

New question
SI-LP-01301-00, p. 25, 45

Name ANSWER KEY

1. [a] [b] [c] ■ ____	35. ■ [b] [c] [d] ____	69. [a] [b] ■ [d] ____
2. [a] ■ [c] [d] ____	36. [a] [b] [c] ■ ____	70. [a] ■ [c] [d] ____
3. [a] [b] ■ [d] ____	37. ■ [b] [c] [d] ____	71. ■ [b] [c] [d] ____
4. [a] ■ [c] [d] ____	38. [a] ■ [c] [d] ____	72. [a] [b] [c] ■ ____
5. [a] [b] [c] ■ ____	39. [a] ■ [c] [d] ____	73. [a] ■ [c] [d] ____
6. [a] [b] [c] ■ ____	40. ■ [b] [c] [d] ____	74. [a] ■ [c] [d] ____
7. ■ [b] [c] [d] ____	41. [a] [b] ■ [d] ____	75. ■ [b] [c] [d] ____
8. [a] [b] ■ [d] ____	42. [a] [b] [c] ■ ____	76. ■ [b] [c] [d] ____
9. [a] [b] ■ [d] ____	43. [a] [b] ■ [d] ____	77. [a] ■ [c] [d] ____
10. ■ [b] [c] [d] ____	44. [a] [b] ■ [d] ____	78. [a] [b] [c] ■ ____
11. ■ [b] [c] [d] ____	45. [a] ■ [c] [d] ____	79. [a] ■ [c] [d] ____
12. [a] [b] ■ [d] ____	46. [a] [b] ■ [d] ____	80. [a] ■ [c] [d] ____
13. ■ [b] [c] [d] ____	47. [a] [b] ■ [d] ____	81. ■ [b] [c] [d] ____
14. [a] [b] ■ [d] ____	48. [a] [b] [c] ■ ____	82. [a] [b] [c] ■ ____
15. ■ [b] [c] [d] ____	49. ■ [b] [c] [d] ____	83. [a] ■ [c] [d] ____
16. [a] [b] ■ [d] ____	50. ■ [b] [c] [d] ____	84. ■ [b] [c] [d] ____
17. [a] ■ [c] [d] ____	51. [a] [b] [c] ■ ____	85. [a] [b] ■ [d] ____
18. ■ [b] [c] [d] ____	52. ■ [b] [c] [d] ____	86. [a] ■ [c] [d] ____
19. [a] [b] ■ [d] ____	53. [a] [b] ■ [d] ____	87. [a] [b] [c] ■ ____
20. ■ [b] [c] [d] ____	54. ■ [b] [c] [d] ____	88. [a] ■ [c] [d] ____
21. [a] ■ [c] [d] ____	55. [a] ■ [c] [d] ____	89. ■ [b] [c] [d] ____
22. [a] [b] ■ [d] ____	56. [a] [b] ■ [d] ____	90. [a] [b] ■ [d] ____
23. ■ [b] [c] [d] ____	57. [a] ■ [c] [d] ____	91. [a] ■ [c] [d] ____
24. ■ [b] [c] [d] ____	58. [a] ■ [c] [d] ____	92. [a] [b] [c] ■ ____
25. [a] [b] ■ [d] ____	59. [a] [b] [c] ■ ____	93. ■ [b] [c] [d] ____
26. [a] [b] ■ [d] ____	60. [a] [b] ■ [d] ____	94. [a] [b] [c] ■ ____
27. [a] [b] [c] ■ ____	61. ■ [b] [c] [d] ____	95. [a] [b] [c] ■ ____
28. [a] [b] ■ [d] ____	62. [a] [b] [c] ■ ____	96. [a] ■ [c] [d] ____
29. [a] ■ [c] [d] ____	63. [a] ■ [c] [d] ____	97. [a] [b] ■ [d] ____
30. [a] [b] [c] ■ ____	64. [a] ■ [c] [d] ____	98. ■ [b] [c] [d] ____
31. [a] [b] ■ [d] ____	65. [a] [b] ■ [d] ____	99. [a] ■ [c] [d] ____
32. [a] ■ [c] [d] ____	66. [a] [b] ■ [d] ____	100. [a] [b] [c] ■ ____
33. [a] [b] [c] ■ ____	67. [a] [b] [c] ■ ____	
34. [a] [b] [c] ■ ____	68. [a] [b] [c] ■ ____	

REFERENCE INDEX

FIGURES SECTION

Control Rod Drive Hydraulic System

Service and Instrument Air Flowpath (Unit 2)

PLANT PROCEDURES SECTION

Attachment 1 of 34AB-B21-002-2S, "RPV Water Level Correction"

Attachment 4 of 73EP-EIP-054-0S, "Protective Action Recommendation to State and Local Authorities"

EOP FLOWCHARTS SECTION

Unit 2 SCC Flowchart (SC/L Path, and Table 5)

Unit 2 CP-3 Flowchart

EOP GRAPHS SECTION

Unit 2 Graph 8, "Drywell Spray Initiation Limit"

Unit 2 Graph 12A, "RHR Pump NPSH Limit"
(Suppression Pool Water Level at or Above 146")

Unit 2 Graph 12B, "RHR Pump NPSH Limit"
(Suppression Pool Water Level Below 146")

TECH SPECS and TRM SECTION

Unit 2 Figure 3.1.7-1, "Sodium Pentaborate Solution Volume Versus Concentration Requirements"

Unit 2 Figure 3.1.7-2, "Sodium Pentaborate Solution Temperature Versus Concentration Requirements"

Unit 2 T 8.0 Secondary Containment

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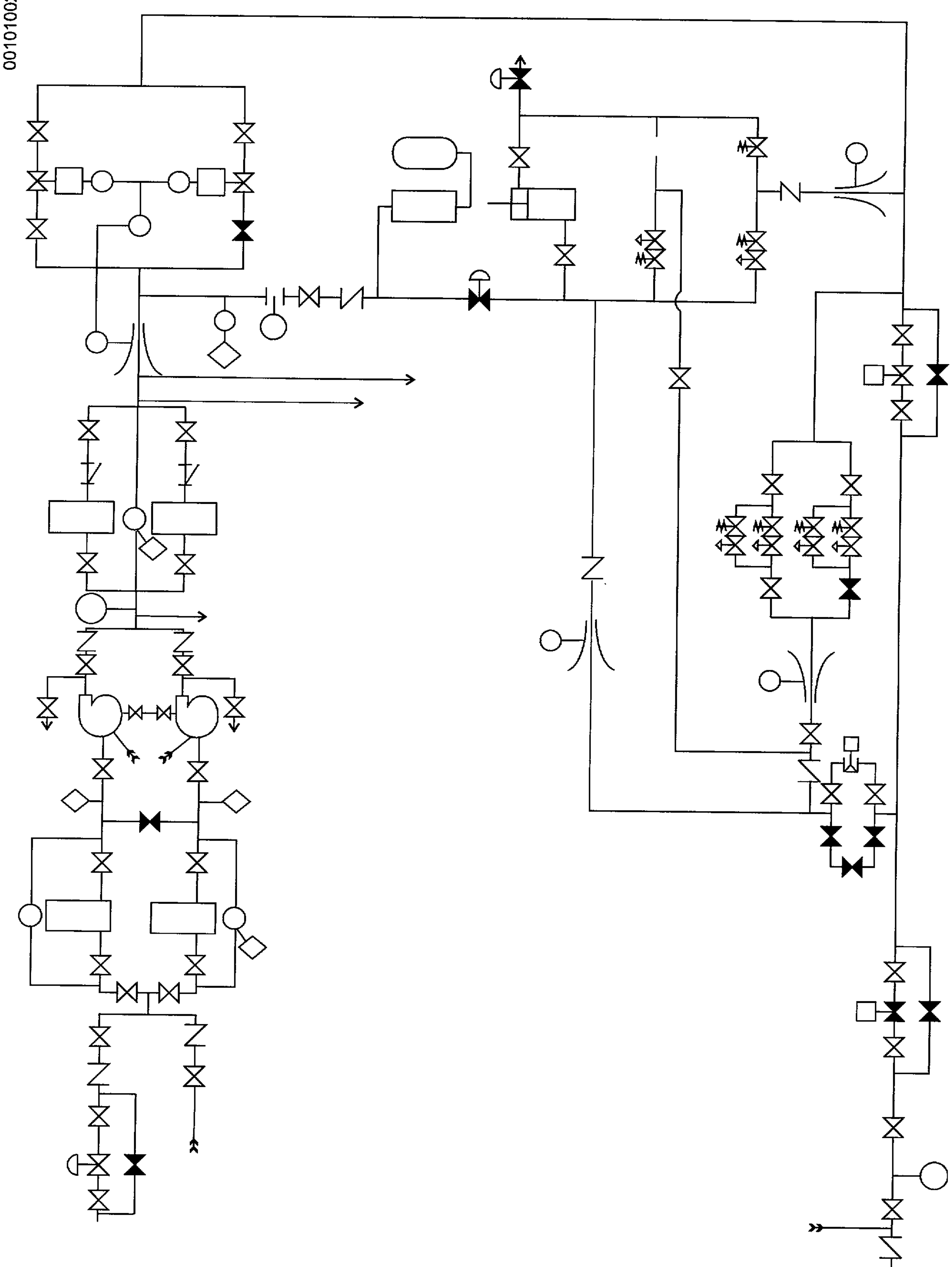
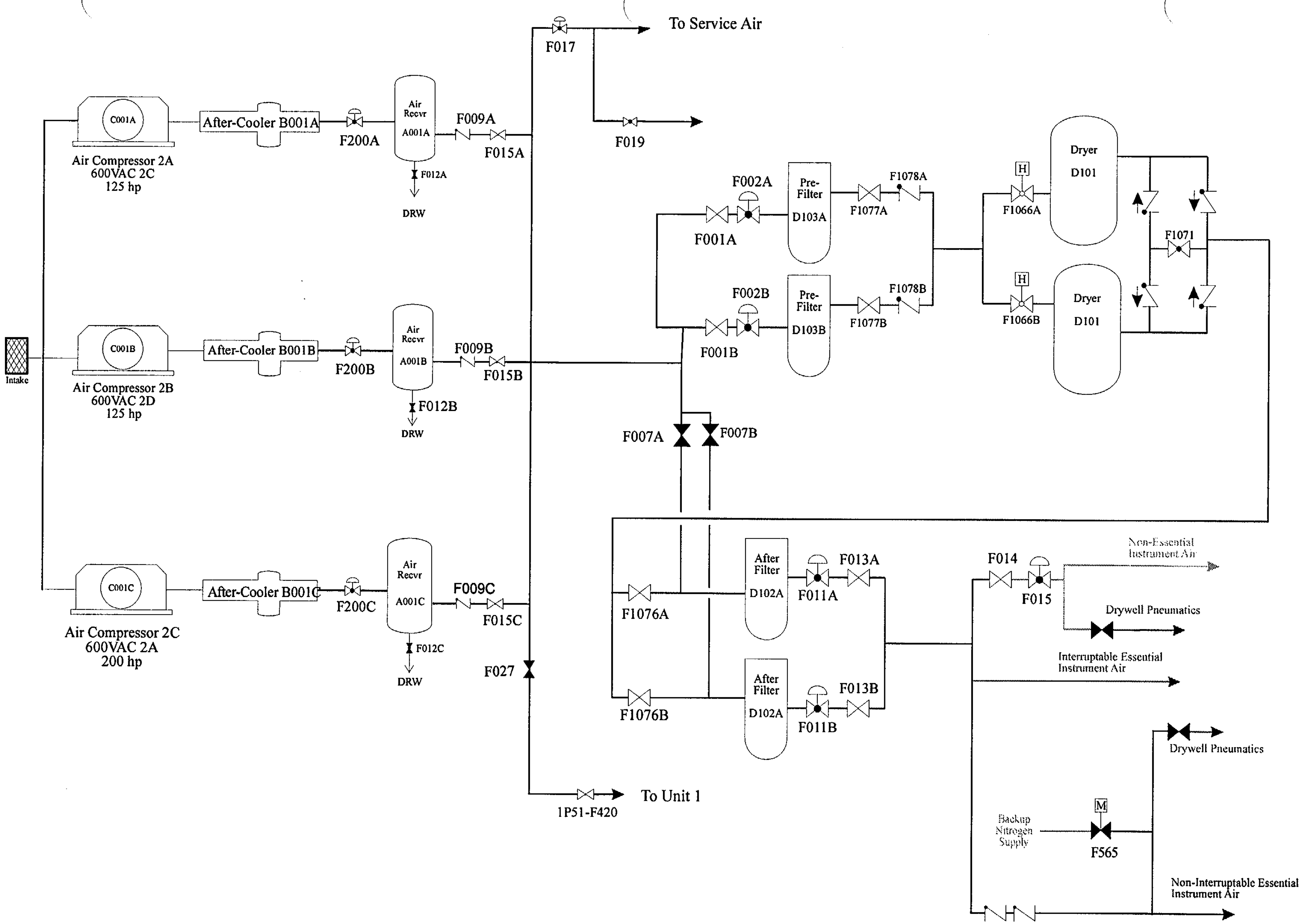


CONTROL ROD DRIVE HYDRAULIC SYSTEM



Service and Instrument Air Flowpath (Unit 2)

DRYWELL RTD GROUPS

RTD	INDICATOR	PANEL
RTD GROUP 1		
2T47-N001K*	2T47-R627, Point 7	2H11-P650
2T47-N010*	2T47-R627, Point 12	2H11-P650
2T47-N014	2T47-R620	2H11-P657
RTD GROUP 2		
2T47-N001A*	2T47-R626, Point 6	2H11-P657
2T47-N002*	2T47-R626, Point 9	2H11-P657
2T47-N015	2T47-R621	2H11-P654
RTD GROUP 3		
2T47-N003*	2T47-R627, Point 9	2H11-P650
2T47-N001M*	2T47-R627, Point 8	2H11-P650
RTD GROUP 4		
2T47-N001L*	2T47-R626, Point 8	2H11-P657
2T47-N009*	2T47-R626, Point 13	2H11-P657

* SPDS Indications may be used for RTD temperature indications

CAUTION 1

RPV WATER LEVEL INSTRUMENT MAY BE USED ONLY WHEN BOTH OF THE FOLLOWING CONDITIONS ARE MET FOR THAT INSTRUMENT:

1. THERE IS NO INDICATION OF ERRATIC INSTRUMENT BEHAVIOR (BOILING COULD OCCUR WHEN THE HIGHEST TEMPERATURE IN THE RTD GROUP IS ABOVE THE RPV SATURATION TEMPERATURE (ATTACHMENT 2 OR SPDS).

   INCREASE INSTRUMENT MONITORING WHEN RPV SATURATION TEMPERATURE FOR INSTRUMENT IS REACHED.

INSTRUMENT	RTD GROUP
NARROW RANGE (0 to +60 IN.)	
- 2C32-R606A & C	1
- 2C32-R606B	2
WIDE RANGE (-150 to +60 IN.)	
- 2B21-R604A & R623A	1
- 2B21-R604B & R623B	2
FLOODUP RANGE (-0 to +400 IN.)	
- 2B21-R605	1 & 2
FLOODUP RANGE (0 to +200 IN.)	
- 2C32-R655	1 & 2
FUEL ZONE RANGE (-317 TO -17 IN.)	
- 2B21-R610	2 & 3
- 2B21-R615	1 & 4

CAUTION 1 (CONTINUED)

2. FOR THE FOLLOWING TABLE, THE WATER LEVEL INSTRUMENT READS ABOVE MINIMUM INDICATED LEVEL FOR THE ASSOCIATED MAXIMUM RUN TEMPERATURE MEASURED BY THE HIGHEST TEMPERATURE IN THE ASSOCIATED RTD GROUP.

a. NARROW RANGE (0 TO +60 IN.)

INSTRUMENT	RTD GROUP
2C32-R606A&C	1
2C32-R606B	2

MINIMUM INDICATED LEVEL (IN.)	MAXIMUM RUN TEMPERATURE (°F)
0	UP TO 273
6	274 TO 350
9	351 TO 399
27	400 OR ABOVE

b. WIDE RANGE (-150 TO +60 IN)

INSTRUMENT	RTD GROUP
2B21-R604A & R623A	1
2B21-R604B & R623B	2

MINIMUM INDICATED LEVEL (IN.)	MAXIMUM RUN TEMPERATURE (°F)
- 150	UP TO 197
- 130	198 TO 350
- 122.5	351 TO 399
- 90	400 OR ABOVE

c. FLOODUP RANGE

INSTRUMENT	RTD GROUP
2B21-R605 (0 TO +400 IN.)	1
2C32-R655 (0 TO +200 IN.)	2

MINIMUM INDICATED LEVEL (IN.)	MAXIMUM RUN TEMPERATURE (°F)
0	UP TO 190
16	191 TO 250
39	251 TO 350
53	351 TO 399
116	400 OR ABOVE

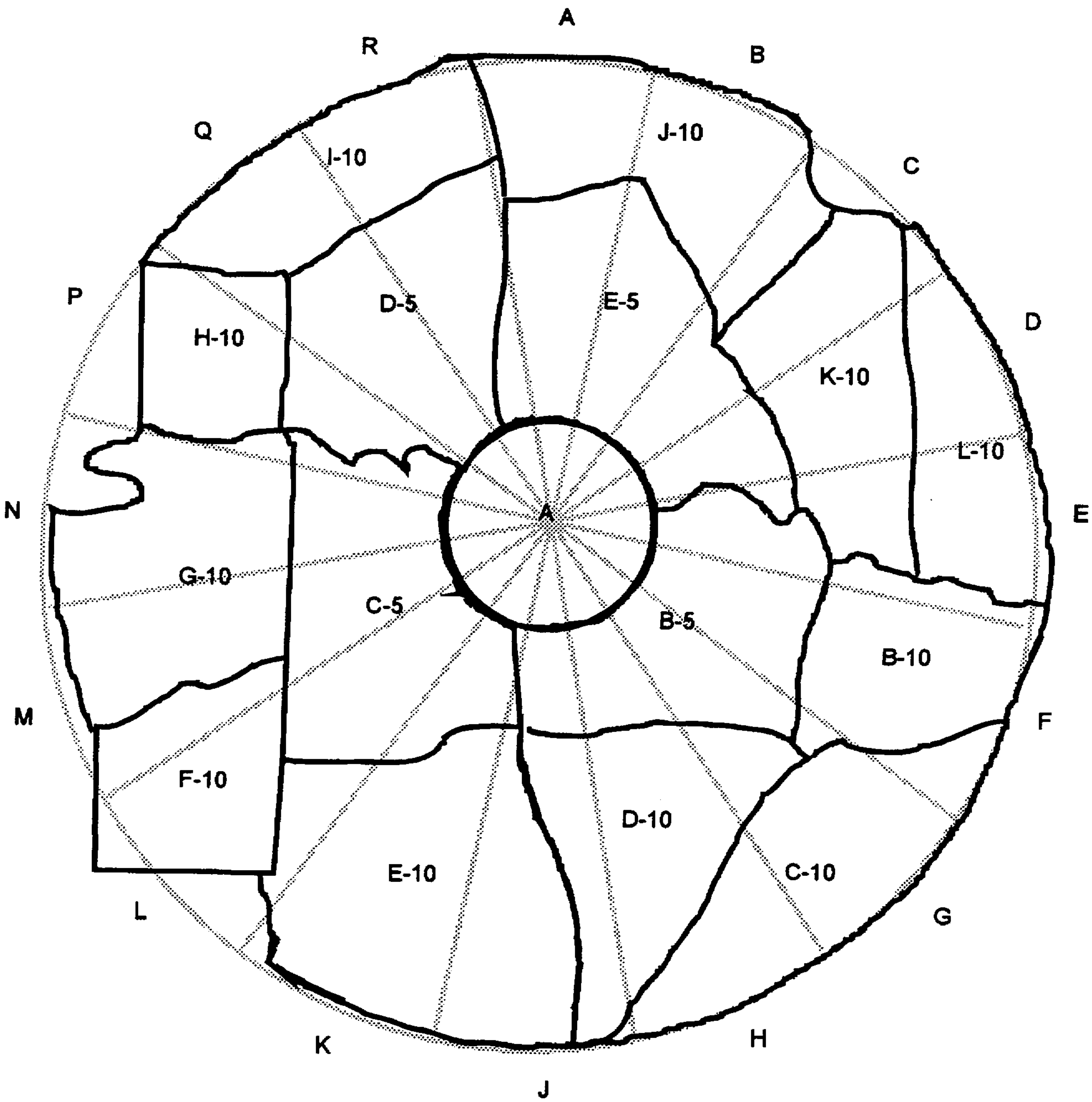
d. FUEL ZONE (-317 TO -17 IN.)

INSTRUMENT	RTD GROUP
2B21-R610	2
2B21-R615	1

MINIMUM INDICATED LEVEL (IN.)	MAXIMUM RUN TEMPERATURE (°F)
- 317	UP TO 280
- 299	281 AND ABOVE

CAUTION 2

2B21-LI-R604A(B) AND 2B21-LR-R623A(B) CANNOT BE USED TO DETERMINE RPV WATER LEVEL DURING RAPID RPV DEPRESSURIZATION BELOW 500 PSIG.



A
B
C
D
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I-10
H-10
D-5
E-5
K-10
L-10
G-10
C-5
B-5
B-10
F-10
E-10
D-10
C-10

SC - SECONDARY CONTAINMENT CONTROL

WHILE PERFORMING THE FOLLOWING

IF PRIMARY CONTAINMENT FLOODING IS OR HAS BEEN REQUIRED THEN exit the EOPs and enter the Severe Accident Guidelines

Go to SAGs

WHILE PERFORMING THE FOLLOWING

IF ANY Unit 1 or Unit 2 secondary containment HVAC exhaust radiation level exceeds the isolation setpoint (Table 14) THEN confirm
- Unit 1 and Unit 2 Reactor Building HVAC isolation
- Unit 1 and Unit 2 Refuel Floor HVAC isolation
- Unit 1 and Unit 2 SBGT initiation

per 34AB-T22-003-2S

IF Refuel Floor HVAC isolates AND a Unit 1 or Unit 2 secondary containment radiation condition does NOT exist THEN restart Refuel Floor HVAC per 34SO-T41-006-2S. If necessary defeat high drywell pressure AND low RWL isolation interlocks per 31EO-EOP-100-2S

IF Reactor Building HVAC isolates AND a Unit 1 or Unit 2 secondary containment radiation condition does NOT exist THEN restart Reactor Building HVAC per 34SO-T41-005-2S. If necessary defeat high drywell pressure AND low RWL isolation interlocks per 31EO-EOP-100-2S

PERFORM CONCURRENTLY

SC/L

WAIT UNTIL ONE of the following is above Maximum Normal Operating Water Level (Table 5):
- A floor drain sump water level
- An area water level

Operate available sump pumps to restore and maintain water level below Maximum Normal Operating Water Level (Table 5)

IF ONE of the following CANNOT be restored and maintained below Maximum Normal Operating Water Level (Table 5):
- ANY floor drain sump water level
- ANY area water level

THEN isolate ALL systems discharging water into sump or area EXCEPT systems required to:
- assure adequate core cooling
- shut down reactor
- suppress fire
- maintain primary containment integrity

PERFORM CONCURRENTLY

WAIT UNTIL primary system is discharging reactor coolant into secondary containment (Table 7)

BEFORE ANY area water level reaches Maximum Safe Operating Water Level (Table 5)

PERFORM CONCURRENTLY RC(A) point A

WAIT UNTIL area water level is above Maximum Safe Operating Water Level in more than one area (Table 5)

EMERGENCY DEPRESS IS REQUIRED

WAIT UNTIL area water level is above Maximum Safe Operating Water Level in more than one area (Table 5)

Shut down reactor per 34GO-OPS-013-2S or 34GO-OPS-014-2S

Table 5 SECONDARY CONTAINMENT OPERATING WATER LEVELS

AREA WATER LEVEL ANNUNCIATORS on 2H11-P657	Max Normal Operating Value	Max Safe Operating Value
SOUTHWEST DIAGONAL AREA		260 in. above 87' el.
1 RB S-W DIAGONAL FLOOR DRN SUMP LEVEL HIGH (657-033) (2T45-N007)	High	
2 CRD S-W DIAG INSTR SUMP LVL HIGH-HIGH-HIGH (657-032) (2T45-N005)	High High High	
SOUTHEAST DIAGONAL AREA		15 in. above 87' el.
3 RB S-E DIAGONAL FLOOR DRN SUMP HIGH-HIGH (657-034) (2T45-N006)	High High	
4 RHR-CS S-E DIAG INSTR SUMP LVL HIGH-HIGH-HIGH (657-103) (2T45-N003B)	High High High	
NORTHWEST DIAGONAL AREA		22 in. above 87' el.
5 RCIC N-W DIAG INSTR SUMP LVL HIGH-HIGH-HIGH (657-014) (2T45-N004)	High High High	
NORTHEAST DIAGONAL AREA		14 in. above 87' el.
6 RHR-CS N-E DIAG INSTR SUMP LVL HIGH-HIGH-HIGH (657-085) (2T45-N003A)	High High High	
HPCI ROOM AREA		14 in. above 87' el.
7 HPCI COMPT INSTR SUMP LEVEL HIGH-HIGH (657-066) (2T45-N001)	High High	
TORUS ROOM AREA		11 in. above 87' el.
8 TORUS N-E AREA INSTR SUMP LVL HIGH-HIGH-HIGH (657-013) (2T45-N002A)	High High High	
9 TORUS S-E AREA INSTR SUMP LVL HIGH-HIGH-HIGH (657-031) (2T45-N002B)	High High High	
10 TORUS N-W AREA INSTR SUMP LVL HIGH-HIGH-HIGH (657-049) (2T45-N002C)	High High High	
11 TORUS S-W AREA INSTR SUMP LVL HIGH-HIGH-HIGH (657-067) (2T45-N002D)	High High High	

CP-3 ATWS LEVEL CONTROL

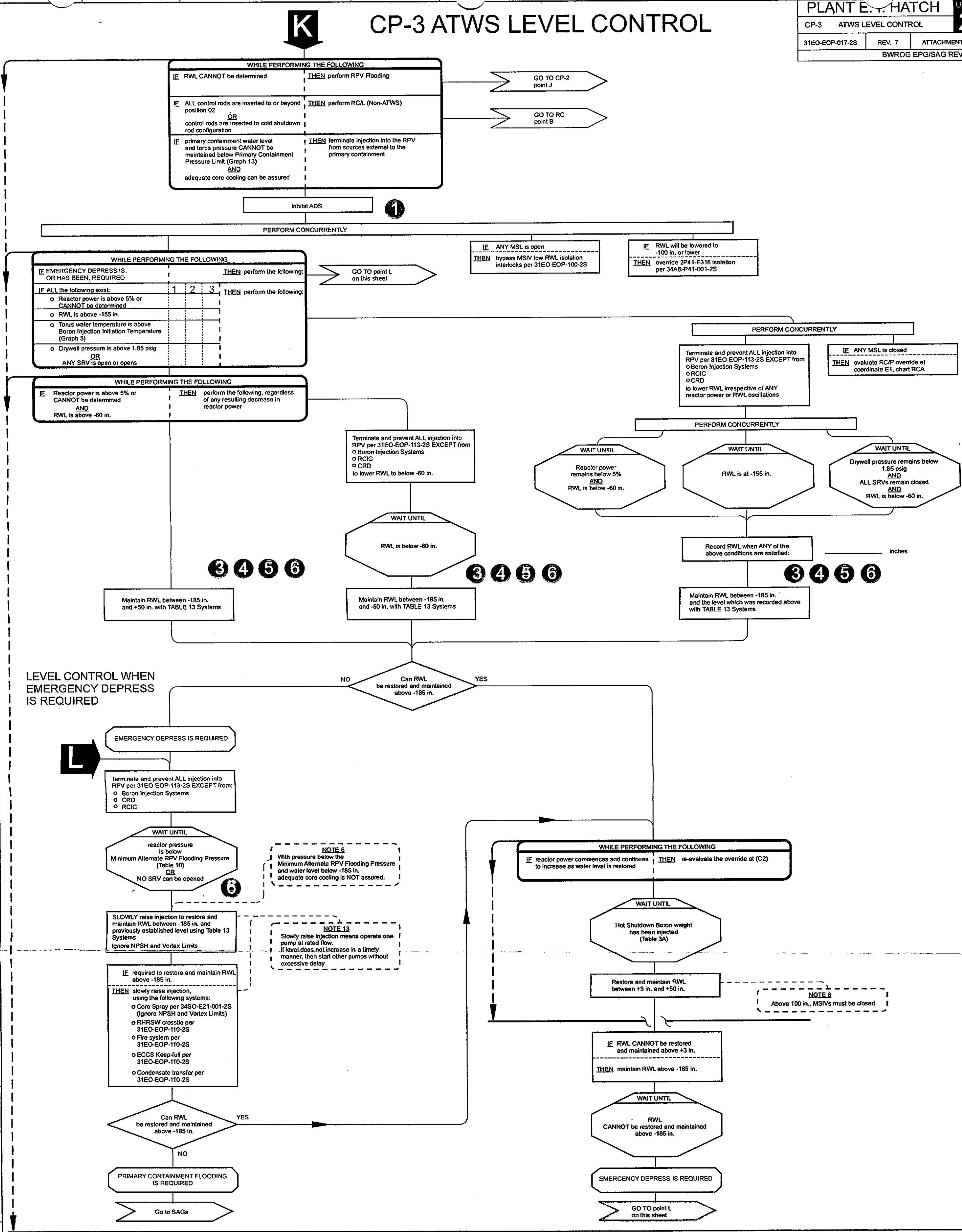


Table 3A HOT SHUTDOWN BORON WEIGHT

	SBLC Tank Level
When using SBLC:	35%
When using alternate boron injection methods see 31EO-EOP-109-2S	

Table 10 MINIMUM ALTERNATE RPV FLOODING PRESSURE

Number of open SRVs	Minimum Alternate RPV Flooding Pressure (psig)
7 or more	100
6	120
5	140
4	180
3	250
2	380
1	760

Table 13 CP-3 INJECTION SYSTEMS 3 4 5 6

- Condensate/feedwater per 34SO-N21-007-2S
- CRD per 34SO-C11-005-2S
- RCIC, with suction from the condensate storage tank if available, per 34SO-E51-001-2S. If necessary, defeat any or all of the following:
  - -high torus water level suction transfer logic per 31EO-EOP-100-2S
  - -low reactor pressure isolation per 31EO-EOP-100-2S
  - -high area temperature isolation per 31EO-EOP-100-2S
- HPCI, with suction from the condensate storage tank if available, per 34SO-E41-001-2S. If necessary, defeat one or both of the following:
  - -high torus water level suction transfer logic per 31EO-EOP-100-2S
  - -high area temperature isolation per 31EO-EOP-100-2S
- LPCI, with injection through the heat exchangers as soon as possible per 34SO-E11-010-2S
- RHR Crosstie per 31EO-EOP-110-2S

1 CAUTION 1

RWL instrument may be used only if BOTH of the following exist:
- there is no indication of erratic instrument behavior
- the instrument reads above minimum level for the associated reference leg maximum run temperature

See 34AB-B21-002-2S for Caution 1

3 CAUTION 3

Operation of HPCI, RCIC, CS, or RHR with suction from the torus and pump flow above the NPSH or vortex limit may result in equipment damage

4 CAUTION 4

Elevated torus pressure may trip the RCIC turbine on high exhaust pressure of 40 psig

5 CAUTION 5

Operation of HPCI or RCIC turbines with suction temperature above 140 °F may result in equipment damage

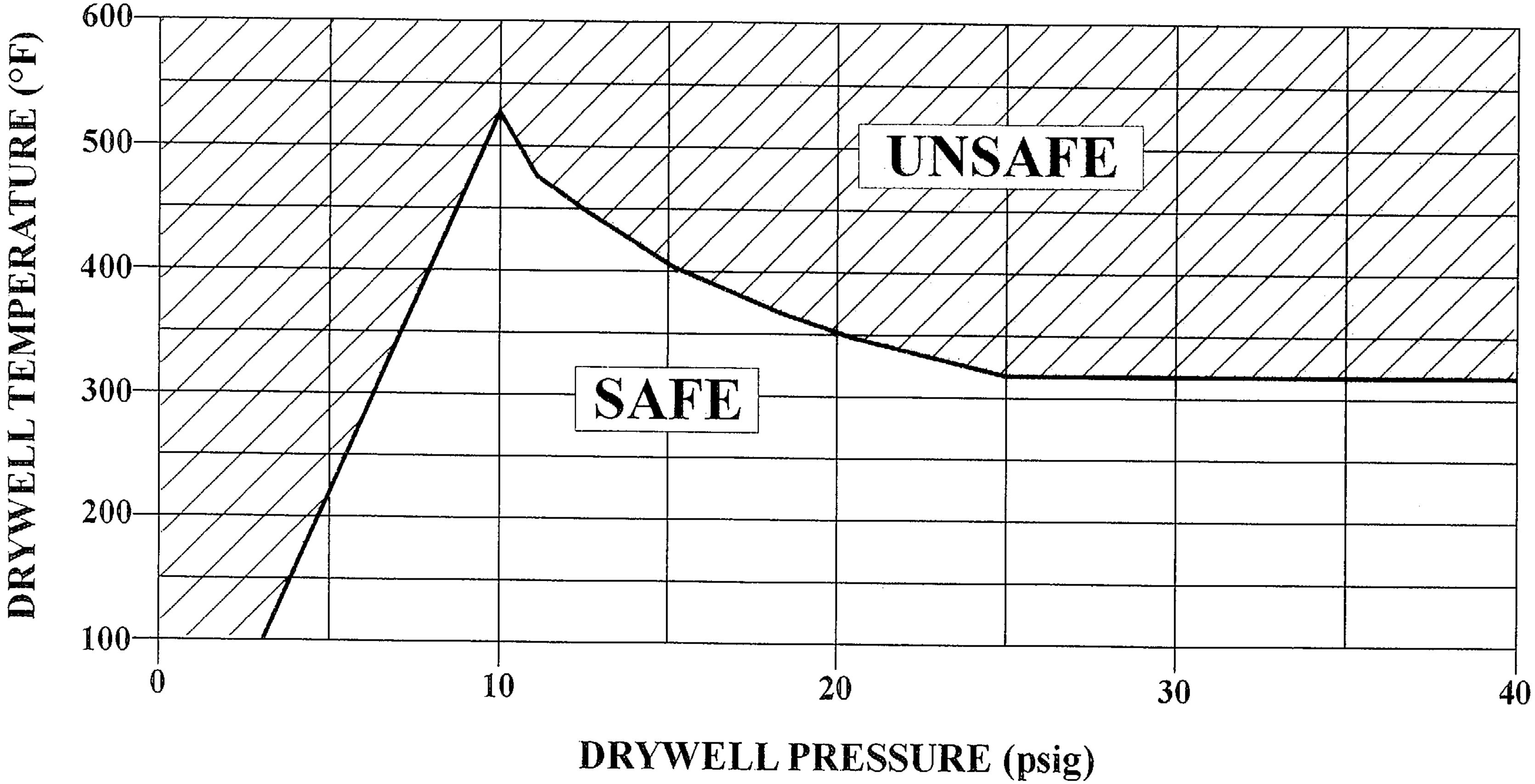
6 CAUTION 6

A rapid rise in injection into the RPV may induce a large power excursion and result in substantial core damage

GRAPH 8

UNIT 2

DRYWELL SPRAY INITIATION LIMIT

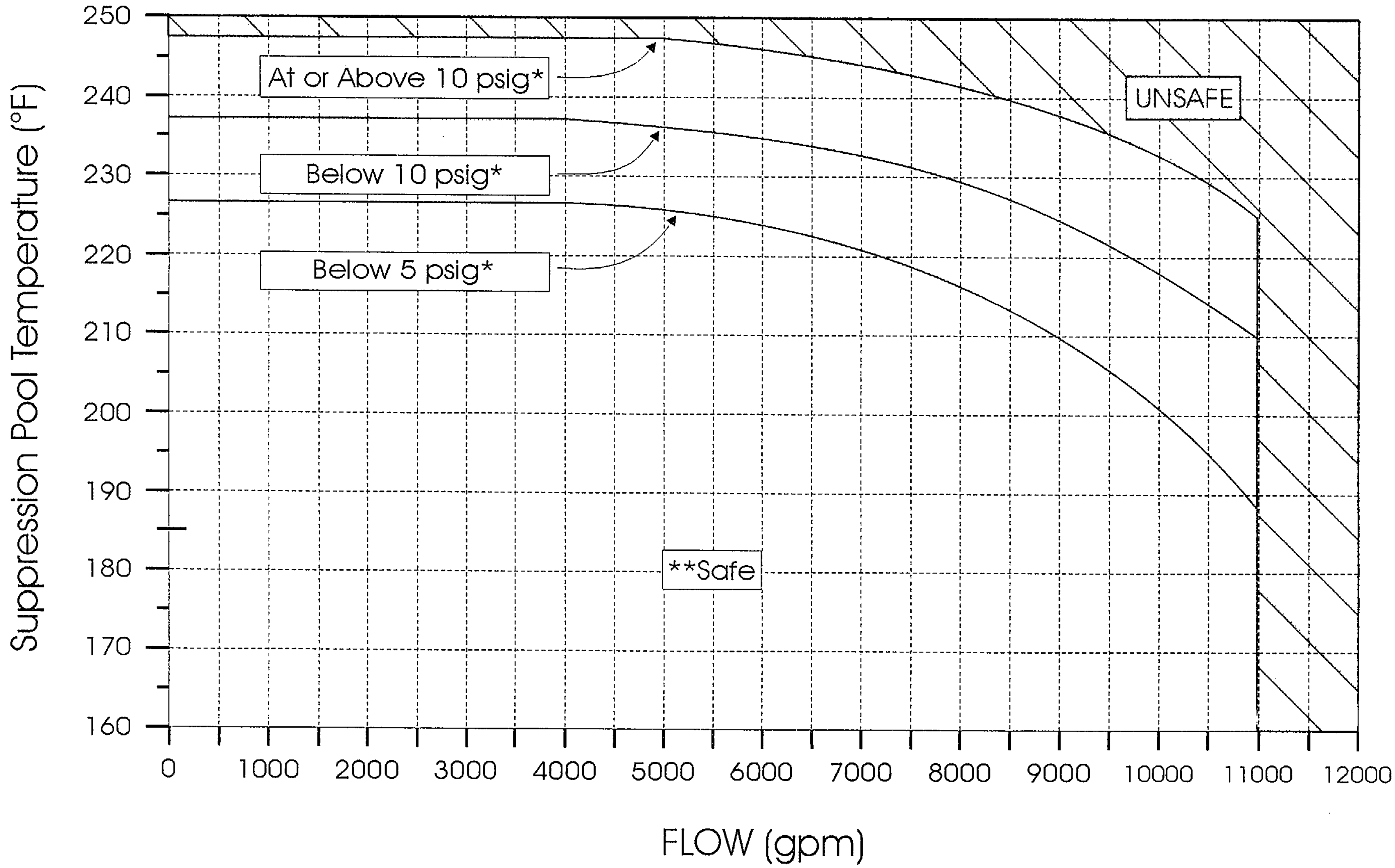


NOTE: May use SPDS Emergency Displays in place of this Graph.

GRAPH 12A

UNIT 2

RHR Pump NPSH Limit
(Suppression Pool Water Level At or Above 146")

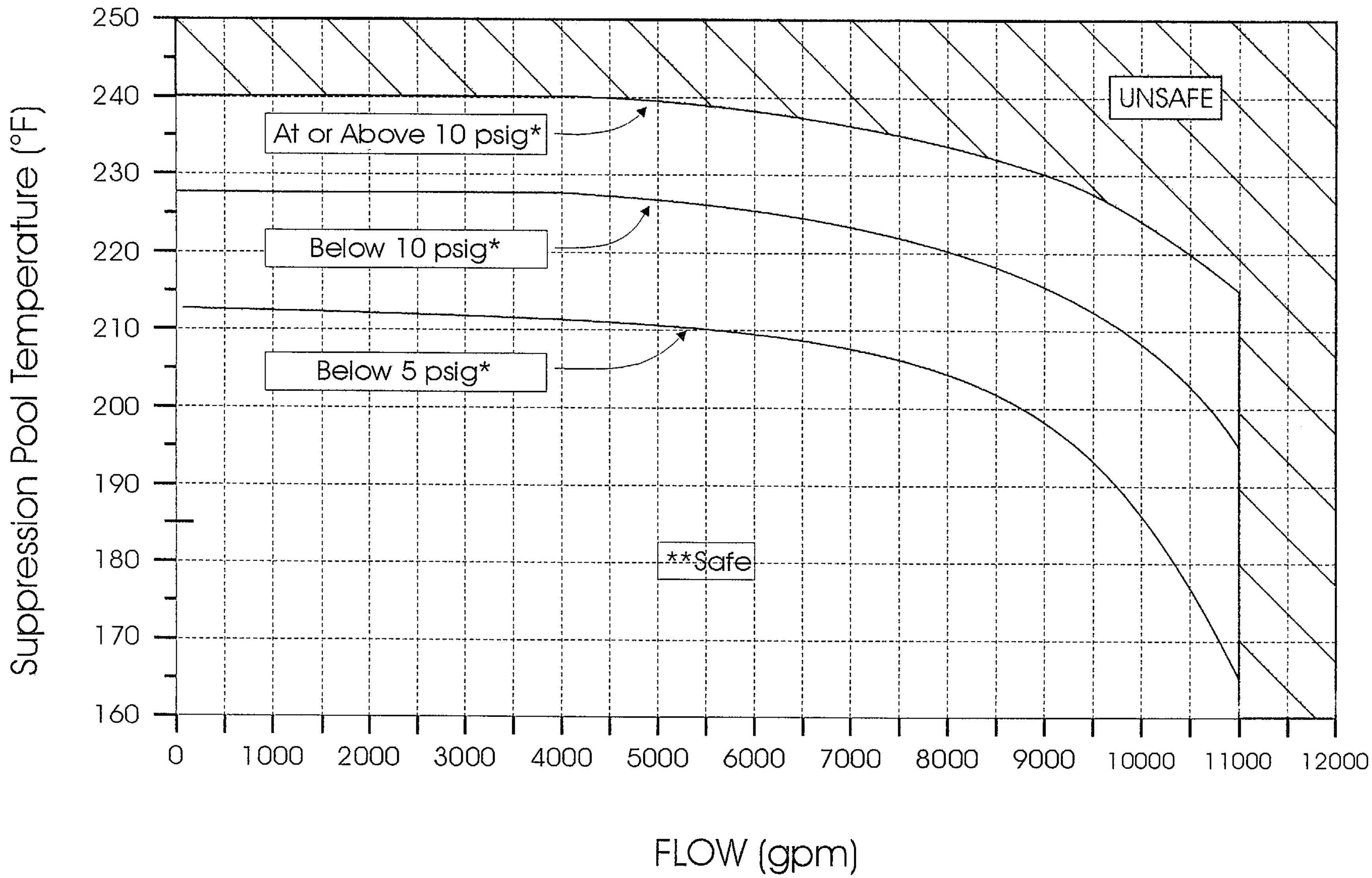


Note: May use SPDS Emergency Displays in place of this Graph.
* Suppression Chamber Pressure.
** Safe operating region is below the applicable pressure line.

GRAPH 12B

RHR Pump NPSH Limit
(Suppression Pool Water Level Below 146")

UNIT 2



Note: May use SPDS Emergency Displays in place of this Graph.

* Suppression Chamber Pressure.

** Safe operating region is below the applicable pressure line.

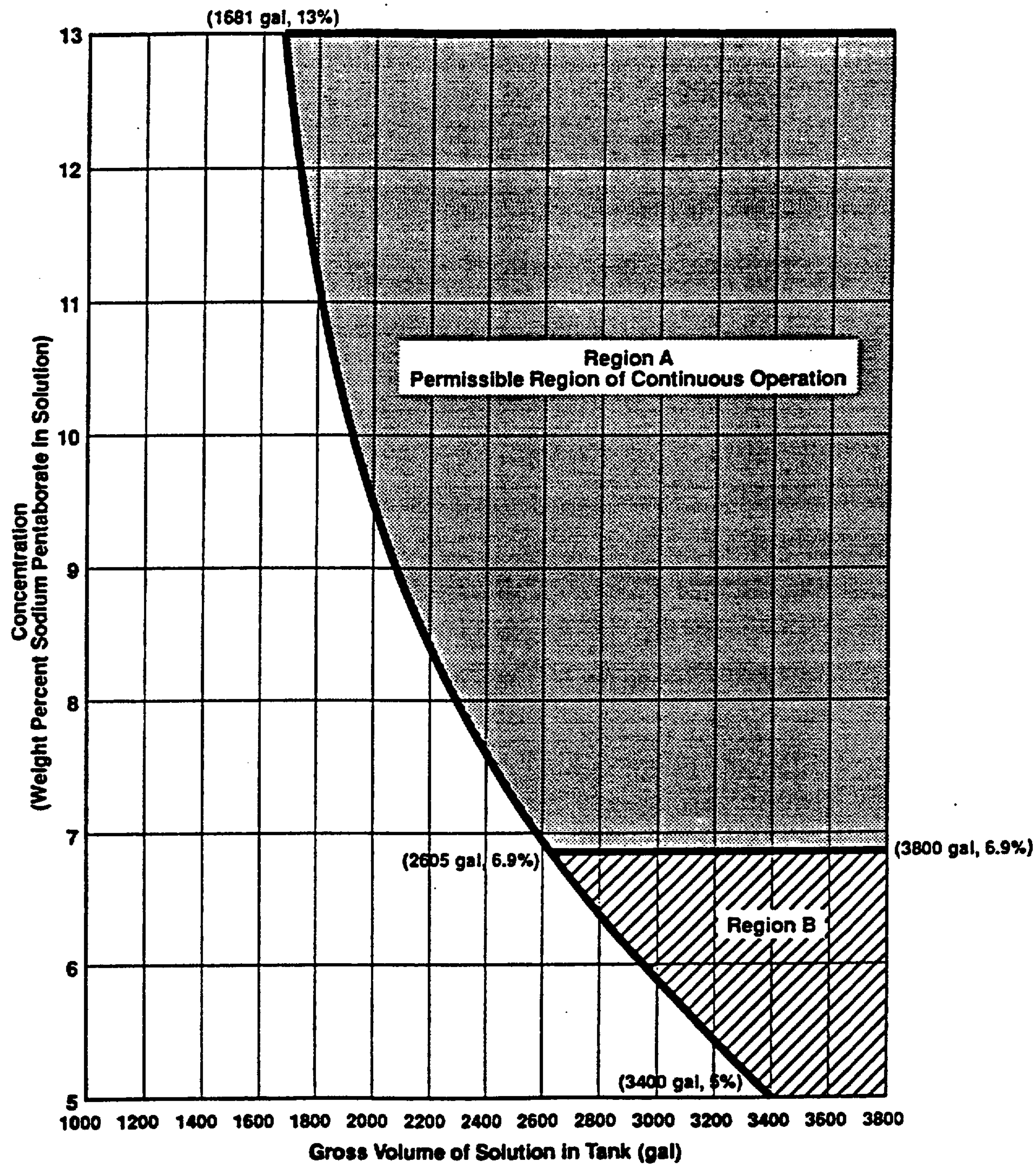


Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

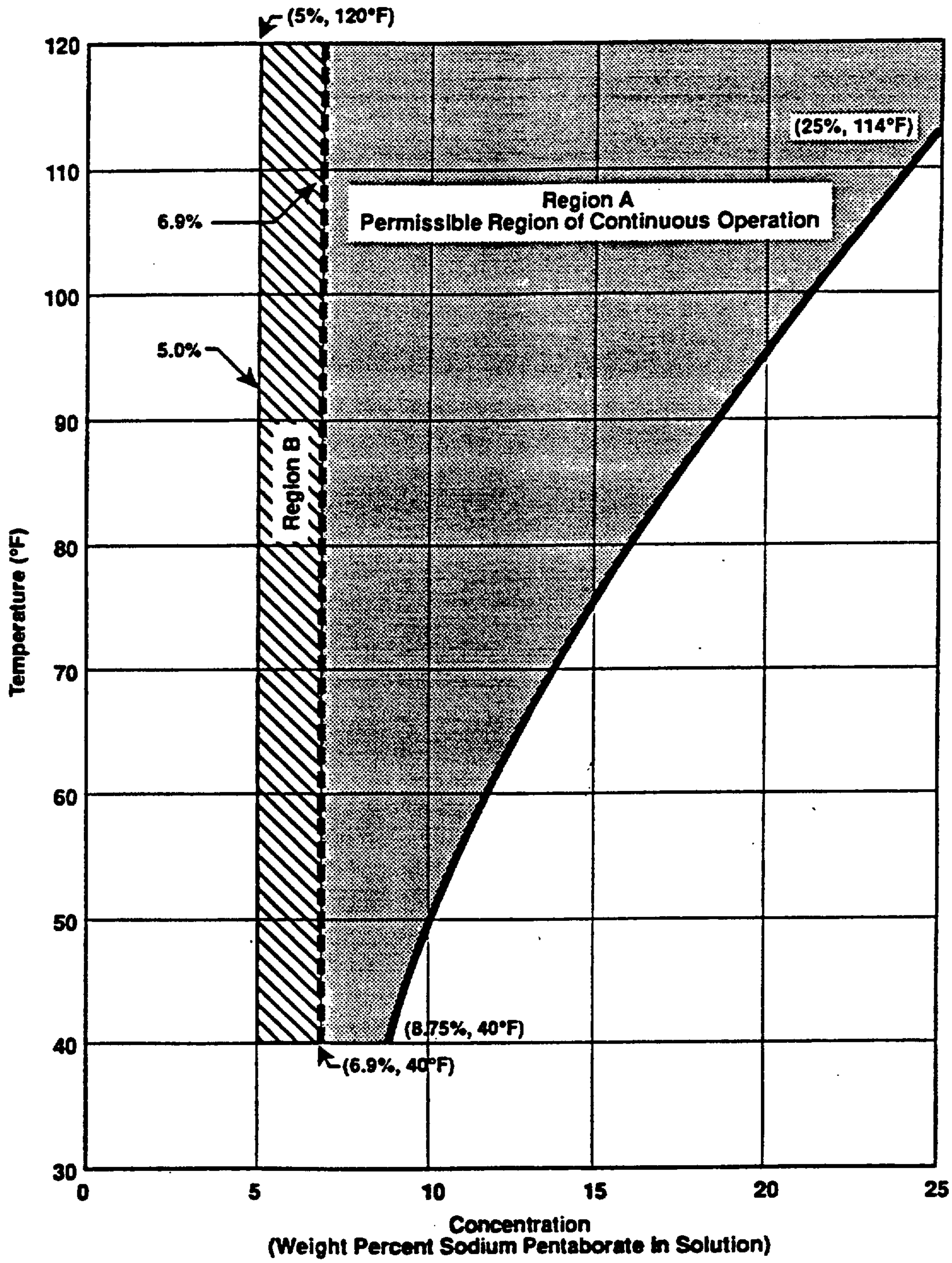
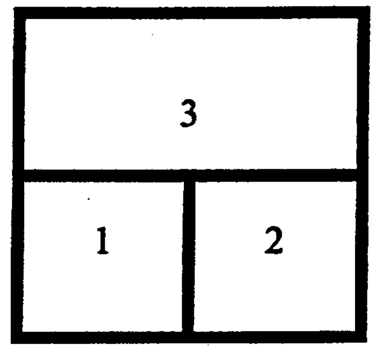


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature
Versus Concentration Requirements

T 8.1 SECONDARY CONTAINMENT TYPES

Type A

Unit	Status Restrictions	R/F Equip. Hatch
1	None	Either IN or OUT
2	None	Either IN or OUT



LCO 3.6.4.1:

Secondary Containment Boundary

Zone 1: Unit 1 Reactor Building
Zone 2: Unit 2 Reactor Building
Zone 3: Common Refueling Floor

Additional Containment Requirements

None

Surveillance Requirements

SR 3.6.4.1.1: Required hatches defined by Table T8.2-1.

SR 3.6.4.1.2: Required doors defined by Table T8.2-1.

SR 3.6.4.1.3 and SR 3.6.4.1.4: 3 SGT subsytems required for surveillances.
1 combination of 3 SGTs to be tested every 18 mo. such that all combinations are tested every 72 (+25%) months.

Required Combination	Last Surveillance Date
1A, 1B, 2A *	
1A, 1B, 2B *	
1A, 2A, 2B	
1B, 2A, 2B	

* One Unit 1 SGT subsystem may trip off per design.

LCO 3.6.4.2

Required Secondary Containment Isolation Valves

As defined by TRM Table T8.2-1.

LCO 3.6.4.3

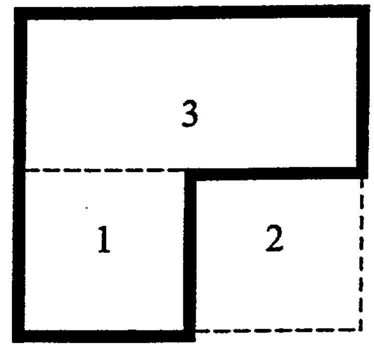
Required Standby Gas Treatment Subsystems

All four: 1A, 1B, 2A, 2B

T 8.1 SECONDARY CONTAINMENT TYPES (Continued)

Type B1

Unit	Status Restrictions	R/F Equip. Hatch
1	None	OUT
2	In MODE 4, 5, or defueled and not conducting OPDRVs.	IN



LCO 3.6.4.1: **Secondary Containment Boundary**

Zone 1: Unit 1 Reactor Building
Zone 3: Common Refueling Floor

Additional Containment Requirements

1. Unit 2 reactor coolant temperature <212°F and vented.
2. No R/F airspace to Unit 2 R/B airspace opening exists via the drywell.

Surveillance Requirements

SR 3.6.4.1.1: Required hatches defined by Table T8.2-1.

SR 3.6.4.1.2: Required doors defined by Table T8.2-1.

SR 3.6.4.1.3 and SR 3.6.4.1.4: 2 (of the 3 required by LCO 3.6.4.3) SGT subsystems required for surveillances.
1 combination of 2 SGTs to be tested every 18 mo. such that all combinations are tested every 90 (+25%) months.

Required Combination	Last Surveillance Date
1A, 2A	
1A, 2B	
1B, 2A	
1B, 2B	
2A, 2B	

LCO 3.6.4.2 **Required Secondary Containment Isolation Valves**

As defined by TRM Table T8.2-1.

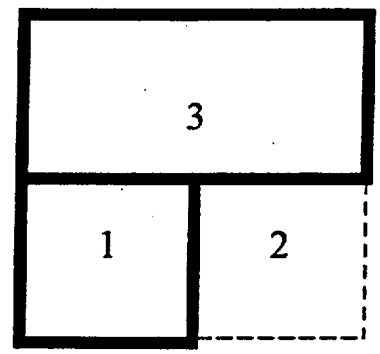
LCO 3.6.4.3 **Required Standby Gas Treatment Subsystems**

One Unit 1 subsystem and both Unit 2 subsystems: 1A, 2A, 2B, or 1B, 2A, 2B

T 8.1 SECONDARY CONTAINMENT TYPES (Continued)

Type B2

Unit	Status Restrictions	R/F Equip. Hatch
1	None	IN
2	In MODE 4, 5, or defueled and not conducting OPDRVs.	IN



LCO 3.6.4.1:

Secondary Containment Boundary

Zone 1: Unit 1 Reactor Building
Zone 3: Common Refueling Floor

Additional Containment Requirements

1. Unit 2 reactor coolant temperature <212°F and vented.
2. No R/F airspace to Unit 2 R/B airspace opening exists via the drywell.

Surveillance Requirements

SR 3.6.4.1.1: Required hatches defined by Table T8.2-1.

SR 3.6.4.1.2: Required doors defined by Table T8.2-1.

SR 3.6.4.1.3 and SR 3.6.4.1.4: 3 SGT subsystems required for surveillances.
1 combination of 3 SGTs to be tested every 18 mo. such that all combinations are tested every 72 (+25%) months

Required Combination	Last Surveillance Date
1A, 1B, 2A*	
1A, 1B, 2B*	
1A, 2A, 2B	
1B, 2A, 2B	

* One Unit 1 SGT subsystem may trip off per design.

LCO 3.6.4.2 **Required Secondary Containment Isolation Valves**

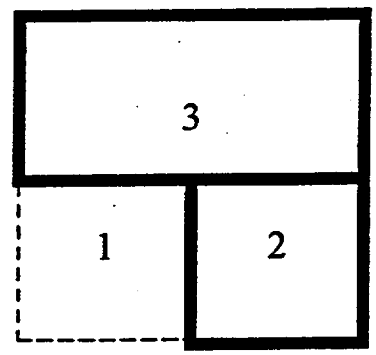
As defined by TRM Table T8.2-1.

LCO 3.6.4.3 **Required Standby Gas Treatment Subsystems**

All four: 1A, 1B, 2A, 2B

Type C

Unit	Status Restrictions	R/F Equip. Hatch
1	In MODE 4, 5, or defueled and not conducting OPDRVs.	IN
2	None	OUT



LCO 3.6.4.1: **Secondary Containment Boundary**

Zone 2: Unit 2 Reactor Building
Zone 3: Common Refueling Floor

Additional Containment Requirements

1. Unit 1 reactor coolant temperature < 212°F and vented.
2. No R/F airspace to Unit 1 R/B airspace opening exists via the drywell.

Surveillance Requirements

SR 3.6.4.1.1: Required hatches defined by Table T8.2-1.

SR 3.6.4.1.2: Required doors defined by Table T8.2-1.

SR 3.6.4.1.3 and SR 3.6.4.1.4: 2 (of the 3 required by LCO 3.6.4.3) SGT subsytems required for surveillances.
1 combination of 2 SGTs to be tested every 18 mo. such that all combinations are tested every 90 (+25%) months.

Required Combination	Last Surveillance Date
2A, 2B	
2A, 1A	
2A, 1B	
2B, 1A	
2B, 1B	

LCO 3.6.4.2 **Required Secondary Containment Isolation Valves**

As defined by TRM Table T8.2-1.

LCO 3.6.4.3 **Required Standby Gas Treatment Subsystems**

Both Unit 2 subsystems and one Unit 1 subsystem:
2A, 2B, 1A or 2A, 2B, 1B

Facility: Hatch (99-301) Date of Examination: 10/29-11/4/99

Item	Task Description	Initials a	Initials b*	Initials c
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model per ES-401.	[illegible]	N/A	mE
	b. Assess whether the outline was systematically prepared and whether all knowledge and ability categories are appropriately sampled.	[illegible]	N/A	mE
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	[illegible]	N/A	mE
	d. Assess whether the repetition from previous examination outlines is excessive.	[illegible]	N/A	mE
2. SIM	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, and major transients.	RLS	RK	[illegible]
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity; ensure each applicant can be tested using at least one new or significantly modified scenario, that no scenarios are duplicated from the applicants' audit test(s)*, and scenarios will not be repeated over successive days.	RLS	RK	[illegible]
	c. To the extent possible, assess whether the outline(s) conform(s) with the qualitative and quantitative criteria specified on Form ES-301-4 and described in Appendix D.	RLS	RK	[illegible]
3. W/T	a. Verify that: (1) the outline(s) contain(s) the required number of control room and in-plant tasks, (2) no more than 30% of the test material is repeated from the last NRC examination, (3)* no tasks are duplicated from the applicants' audit test(s), and (4) no more than 80% of any operating test is taken directly from the licensee's exam banks.	RLS	RK	[illegible]
	b. Verify that: (1) the tasks are distributed among the safety function groupings as specified in ES-301, (2) one task is conducted in a low-power or shutdown condition, (3) 40% of the tasks require the applicant to implement an alternate path procedure, (4) one in-plant task tests the applicant's response to an emergency or abnormal condition, and (5) the in-plant walk-through requires the applicant to enter the RCA.	RLS	RK	[illegible]
	c. Verify that the required administrative topics are covered, with emphasis on performance-based activities.	RLS	RK	[illegible]
	d. Determine if there are enough different outlines to test the projected number and mix of applicants and ensure that no items are duplicated on successive days.	RLS	RK	[illegible]
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam section.	RLS	RK	[illegible]
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.**	RLS	RK	[illegible]
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	[illegible]	N/A	mE
	d. Check for duplication and overlap among exam sections.**	RLS	RK	[illegible]
	e. Check the entire exam for balance of coverage.**	RLS	RK	[illegible]
	f. Assess whether the exam fits the appropriate job level (RO or SRO).**	RLS	RK	[illegible]

	Printed Name / Signature	Date
a. Author	R.L. Smith / [signature] D.C. PAYNE / [signature] (1a-d, 4c)	10/21/99 10/22/99
b. Facility Reviewer(*)	R.S. Grantham / [signature]	10/21/99
c. Chief Examiner	D.C. PAYNE / [signature]	10/22/99
d. NRC Supervisor	ME ERNSTES / [signature]	10/28/99
	T.O. CHRISTENSEN / [signature]	10/29/99

(*) Not applicable for NRC-developed examinations.

** Facility Checked for Balance + 55.45 on the operating part of the Exam only.

Facility: HATCH (99-301) Date of Examination: 11/1 - 4/99
Examination Level (circle one): ~~RO~~ / **SRO** Operating Test Number: 1 & 2

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Technical Specifications (TRM)	JPM: 25033 - Determine Fire Protection Requirements
	Conduct of Operations	JPM: 25032 - Determine Overtime Availability
A.2	Equipment Operability Requirements	JPM: 25034 - Review of Core Spray Vlv Operability Surveillance
A.3	Radiation Areas, Signs, Dose Rates,Stay Time	Question 1: Knowledge of radiation areas, signs, and administrative requirements.
		Question 2: Calculate dose received for a given stay time and level authority required to enter radiation area of 6 R/hr.
A.4	Emergency Plan	Question 1: Knowledge of E-Plan based on contaminated, injured worker during steam leak.
		Question 2: Knowledge of E-Plan based on unisolable HPCI steam line leak with rising Secondary Containment temperatures and rising area radiation levels.

Facility: HATCH (99-301) Date of Examination: 11/1 - 4/99

Exam Level (circle one): ~~RO~~ / **SRO(I)** / ~~SRO(U)~~ Operating Test No.: 1

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
a. 25031 - Move Control Rods Using Single Notch (Rod Drift)	D, A, S	1
b. 25018 - Shutdown HPCI (Normal) (Min. Flow Vlv Failure)	D, A, S	4
c. 28.16M - Perform an EDG Manual Start Surveillance (Lube Oil Line Leak)	M, A, S	6
d. 12.01M - Verify Overlap Between IRM Ranges 6 and 7 with one IRM Out of Service and two IRMs Out of Spec.	M, L, S	7
e. 14.01 - Perform an MSIV Trip Test	D, S	3
f. 34.12 - Restore and Maintain Level within a Specified Range Using RHR Service Water	D, S	2
g. 13.58- Purge Suppression Chamber with Air for H_2 Control	D, A, C	5

B.2 Facility Walk-Through

a. 36.23 - Line Up & Operate Fire System via Condensate Transfer/Shutdown Cooling for RPV Injection	D, L, O, R, E	2
b. 35.02 - Start an Idle Station Service Air Compressor	N, O, R	8
c. 30.07 - From Outside the MCR, During a MCR Evacuation, Locally Start the SBGT System	D, O, R	9

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA, (O)utside, (E)mergency

Facility: HATCH (99-301) Date of Examination: 11/1 - 4/99

Exam Level (circle one): ~~RO~~ / ~~SRO(I)~~ / **SRO(U)** Operating Test No.: 2

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
a. 25031 - Move Control Rods Using Single Notch (Rod Drift)	D, A, S	1
b. 28.16M - Perform an EDG Manual Start Surveillance (Lube Oil Line Leak)	M, A, S	6
c. 13.58- Purge Suppression Chamber with Air for H_2 Control	D, A, C	5
d.		
e.		
f.		
g.		

B.2 Facility Walk-Through

System / JPM Title	Type Code*	Safety Function
a. 36.23 - Line Up & Operate Fire System via Condensate Transfer/Shutdown Cooling for RPV Injection	D, L, O, R, E	2
b. 35.02 - Start an Idle Station Service Air Compressor	N, O, R	8
c.		

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA, (O)utside, (E)mergency

Facility: Hatch (99-301) Date of Examination: 11/1-4/99 Operating Test Number: 1

1. GENERAL CRITERIA	Initials a	Initials b	Initials c
a. The operating test conforms with the previously approved outline; changes are consistent with sampling requirements (e.g., 10 CFR 55.45, operational importance, safety function distribution).	RLS	RSG	[illegible]
b. There is no day-to-day repetition between this and other operating tests to be administered during this examination.	RLS	RSG	[illegible]
c. The operating test shall not duplicate items from the applicants' audit test(s)(see Section D.1.a). Note 1	RLS	RSG	[illegible]
d. Overlap with the written examination and between operating test categories is within acceptable limits.	RLS	RSG	[illegible]
e. It appears that the operating test will differentiate between competent and less-than-competent applicants at the designated license level.	RLS	RSG	[illegible]
2. WALK-THROUGH (CATEGORY A & B) CRITERIA	-	-	-
a. Each JPM includes the following, as applicable: · initial conditions · initiating cues · references and tools, including associated procedures · validated time limits (average time allowed for completion) and specific designation if deemed to be time critical by the facility licensee · specific performance criteria that include: - detailed expected actions with exact criteria and nomenclature - system response and other examiner cues - statements describing important observations to be made by the applicant - criteria for successful completion of the task - identification of critical steps and their associated performance standards - restrictions on the sequence of steps, if applicable	RLS	RSG	[illegible]
b. The prescripted questions in Category A are predominantly open reference and meet the criteria in Attachment 1 of ES-301.	RLS	RSG	[illegible]
c. Repetition from operating tests used during the previous licensing examination is within acceptable limits (30% for the walk-through) and do not compromise test integrity.	RLS	RSG	[illegible]
d. At least 20 percent of the JPMs on each test are new or significantly modified.	RLS	RSG	[illegible]
3. SIMULATOR (CATEGORY C) CRITERIA	-	-	-
a. The associated simulator operating tests (scenario sets) have been reviewed in accordance with Form ES-301-4 and a copy is attached.	RLS	RSG	[illegible]

	Printed Name / Signature	Date
a. Author	R. L. Smith / R. L. Smith	9/9/99
b. Facility Reviewer(*)	R. S. Grantham / R S Grantham	9/9/99
NRC Reviewer	R. F. Aiello	10/1/99
c. NRC Chief Examiner (*)	D. C. Payne / D Payne	10/25/99
d. NRC Supervisor (*)	HO CHRISTENSEN / [illegible]	10/25/99

(*) The facility signature is not applicable for NRC-developed tests; two independent NRC reviews are required.

Note 1: Scenarios are new development - developed independently of audit. JPMs do not overlap audit.

Facility: Hatch (99-301) Date of Exam: 11/1-4/99 Scenario Numbers: 1/2/3 Operating Test No.: 1

	QUALITATIVE ATTRIBUTES	Initials a	Initials b	Initials c
1.	The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue the operators into expected events.	QN	RK	[illegible]
2.	The scenarios consist mostly of related events.	QN	RK	[illegible]
3.	Each event description consists of • the point in the scenario when it is to be initiated • the malfunction(s) that are entered to initiate the event • the symptoms/cues that will be visible to the crew • the expected operator actions (by shift position) • the event termination point (if applicable)	QN	RK	[illegible]
4.	No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.	QN	RK	[illegible]
5.	The events are valid with regard to physics and thermodynamics.	QN	RK	[illegible]
6.	Sequencing and timing of events is reasonable, and allows the examination team to obtain complete evaluation results commensurate with the scenario objectives.	QN	RK	[illegible]
7.	If time compression techniques are used, the scenario summary clearly so indicates. Operators have sufficient time to carry out expected activities without undue time constraints. Cues are given.	QN	RK	[illegible]
8.	The simulator modeling is not altered.	QN	RK	[illegible]
9.	The scenarios have been validated. Any open simulator performance deficiencies have been evaluated to ensure that functional fidelity is maintained while running the planned scenarios.	QN	RK	[illegible]
10.	Every operator will be evaluated using at least one new or significantly modified scenario. All other scenarios have been altered in accordance with Section D.4 of ES-301.	QN	RK	[illegible]
11.	All individual operator competencies can be evaluated, as verified using Form ES-301-6 (submit the form along with the simulator scenarios).	QN	RK	[illegible]
12.	Each applicant will be significantly involved in the minimum number of transients and events specified on Form ES-301-5 (submit the form with the simulator scenarios).	QN	RK	[illegible]
13.	The level of difficulty is appropriate to support licensing decisions for each crew position.	QN	RK	[illegible]

	TARGET QUANTITATIVE ATTRIBUTES (PER SCENARIO; SEE SECTION D.4.D)		Actual Attributes	–	–	–
1.	Total malfunctions (5-8)	See scenarios	/ / / /	QN	RK	[illegible]
2.	Malfunctions after EOP entry (1-2)		/ / / /	QN	RK	[illegible]
3.	Abnormal events (2-4)		/ / / /	QN	RK	[illegible]
4.	Major transients (1-2)		/ / / /	QN	RK	[illegible]
5.	EOPs entered/requiring substantive actions (1-2)		/ / / /	QN	RK	[illegible]
6.	EOP contingencies requiring substantive actions (0-2)		/ / / /	QN	RK	[illegible]
7.	Critical tasks (2-3)		/ / / /	QN	RK	[illegible]

SRO-I #1

OPERATING TEST NO.: 1

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO		Reactivity	1				
		Normal	1				
		Instrument	2				
		Component	2				
		Major	1				
SRO-I	As RO	Reactivity	1		3		
		Normal	0		1		
		Instrument	1		2		
		Component	1		3/4		
		Major	1		5		
	As SRO	Reactivity	0	1			
		Normal	1	1			
		Instrument	1	2			
		Component	1	3			
		Major	1	4			
SRO-U		Reactivity	0				
		Normal	1				
		Instrument	1				
		Component	1				
		Major	1				

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: R. L. Smith

Chief Examiner: D. Payne

SRO-I #2

OPERATING TEST NO.: 1

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO		Reactivity	1				
		Normal	1				
		Instrument	2				
		Component	2				
		Major	1				
SRO-I	As RO	Reactivity	1		3		
		Normal	0		1		
		Instrument	1		2		
		Component	1		3/4		
		Major	1		5		
	As SRO	Reactivity	0	1			
		Normal	1	1			
		Instrument	1	2			
		Component	1	3			
		Major	1	4			
SRO-U		Reactivity	0				
		Normal	1				
		Instrument	1				
		Component	1				
		Major	1				

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: R.L. Smith

Chief Examiner: M Bayne

SRO-I #3

OPERATING TEST NO.: 1

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO		Reactivity	1				
		Normal	1				
		Instrument	2				
		Component	2				
		Major	1				
SRO-I	As RO	Reactivity	1		3		
		Normal	0	1			
		Instrument	1		2		
		Component	1	3	4		
		Major	1	4	5		
	As SRO	Reactivity	0			2	
		Normal	1			1	
		Instrument	1			3	
		Component	1			4	
		Major	1			5	
SRO-U		Reactivity	0				
		Normal	1				
		Instrument	1				
		Component	1				
		Major	1				

Instructions:
(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.
(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: R L Smith

Chief Examiner: [illegible]

SRO-I #4

OPERATING TEST NO.: 1

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO		Reactivity	1				
		Normal	1				
		Instrument	2				
		Component	2				
		Major	1				
SRO-I	As RO	Reactivity	1	1			
		Normal	0			1	
		Instrument	1	2			
		Component	1	3		4	
		Major	1	4		5	
	As SRO	Reactivity	0		1		
		Normal	1		1		
		Instrument	1		2		
		Component	1		3/4		
		Major	1		5		
SRO-U		Reactivity	0				
		Normal	1				
		Instrument	1				
		Component	1				
		Major	1				

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.
(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: R. L. Smith

Chief Examiner: M Payne

SRO-I #5

OPERATING TEST NO.: 1

Applicant Type	Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO	Reactivity	1				
	Normal	1				
	Instrument	2				
	Component	2				
	Major	1				
SRO-I As RO	Reactivity	1			2	
	Normal	0		1		
	Instrument	1			3	
	Component	1		3/4	4	
	Major	1		5	5	
SRO-I As SRO	Reactivity	0	2			
	Normal	1	1			
	Instrument	1	2			
	Component	1	3			
	Major	1	4			
SRO-U	Reactivity	0				
	Normal	1				
	Instrument	1				
	Component	1				
	Major	1				

Instructions:
(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.
(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: R. L. Smith

Chief Examiner: [signature]

SRO-U #1

OPERATING TEST NO.: 1

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO		Reactivity	1				
		Normal	1				
		Instrument	2				
		Component	2				
		Major	1				
SRO-I	As RO	Reactivity	1	1			
		Normal	0	1			
		Instrument	1	2			
		Component	1	3			
		Major	1	4			
	As SRO	Reactivity	0				
		Normal	1				
		Instrument	1				
		Component	1				
		Major	1				
SRO-U		Reactivity	0		3		
		Normal	1		1		
		Instrument	1		2		
		Component	1		3/4		
		Major	1		5		

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: R. J. Smith

Chief Examiner: [signature]

SRO-U #2

OPERATING TEST NO.: 1

Applicant Type		Evolution Type	Minimum Number	Scenario Number 1	2	3	4
RO		Reactivity	1				
		Normal	1				
		Instrument	2				
		Component	2				
		Major	1				
SRO-I	As RO	Reactivity	1	1			
		Normal	0	1			
		Instrument	1	2			
		Component	1	3			
		Major	1	4			
	As SRO	Reactivity	0				
		Normal	1				
		Instrument	1				
		Component	1				
		Major	1				
SRO-U		Reactivity	0		3		
		Normal	1		1		
		Instrument	1		2		
		Component	1		3/4		
		Major	1		5		

Instructions:

(1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: ______________________

Chief Examiner: ______________________

Competencies	SRO-I #1/2 Applicant #1 RO/SRO-I/SRO-U SCENARIO				SRO-I #3 Applicant #2 RO/SRO-I/SRO-U SCENARIO				SRO-I #4 Applicant #3 RO/SRO-I/SRO-U SCENARIO			
	1	2	3	4	1	2	3	4	1	2	3	4
Understand and Interpret Annunciators and Alarms		2/3/4/5			3/4	2/4/5			2/3/4		3/4/5	
Diagnose Events and Conditions		2/3/4/5			3/4	2/4/5			2/3/4		3/4/5	
Understand Plant and System Response		2/3/4/5			3/4	2/4/5			2/3/4		3/4/5	
Comply With and Use Procedures (1)	3/4	2/3/4/5			3/4/1	2/3/4/5	2/3/4/5		1/2/3/4	2/3/4/5	1/3/4/5	
Operate Control Boards (2)		1/3/5			1/3/4	2/3/4/5			1/2/3/4		1/3/4/5	
Communicate and Interact With the Crew	2/3/4	2/3/4/5			2/3/4	2/3/4/5	2/3/4/5		1/2/3/4	2/3/4/5	1/3/4/5	
Demonstrate Supervisory Ability (3)	2/3/4						2/3/4/5			2/3/4/5		
Comply With and Use Tech. Specs. (3)	2						4			2/4		

Notes:

(1) Includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.

Instructions:

Circle the applicant's license type and enter one or more event numbers that will allow the examiners to evaluate every applicable competency for every applicant.

Author: R. L. Smith
Chief Examiner: [illegible]

Competencies	Applicant #1 RO/SRO-I/SRO-U SRO-I #5				Applicant #2 RO/SRO-I/SRO-U SRO-U #1+2				Applicant #3 RO/SRO-I/SRO-U			
	SCENARIO				SCENARIO				SCENARIO			
	1	2	3	4	1	2	3	4	1	2	3	4
Understand and Interpret Annunciators and Alarms		3/4/5	3/4/5		2/3/4							
Diagnose Events and Conditions		3/4/5	3/4/5		2/3/4							
Understand Plant and System Response		3/4/5	2/3/4/5		2/3/4							
Comply With and Use Procedures (1)	2/3/4/5	1/3/4/5	2/3/4/5		1/2/3/4	2/3/4/5						
Operate Control Boards (2)		1/3/4/5	2/3/4/5		1/2/3/4							
Communicate and Interact With the Crew	2/3/4/5	2/3/4/5	2/3/4/5		1/2/3/4	2/3/4/5						
Demonstrate Supervisory Ability (3)	2/3/4/5					2/3/4/5						
Comply With and Use Tech. Specs. (3)	4					4						

Notes:

(1) Includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.

Instructions:

Circle the applicant's license type and enter one or more event numbers that will allow the examiners to evaluate every applicable competency for every applicant.

Author: [signature]
Chief Examiner: [signature]

Facility: Hatch (99-301) Date of Exam: 10/29/99 Exam Level: SRO

Tier	Group	K/A Category Points											Point Total
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	
1. Emergency & Abnormal Plant Evolutions	1	4	4	4				5	4			5	26
	2	1	4	4				3	2			3	17
	Tier Totals	5	8	8				8	6			8	43
2. Plant Systems	1	3	1	2	3	2	1	2	3	1	2	3	23
	2	1	1	2	0	2	0	1	2	2	2	0	13
	3	1	0	0	1	0	1	1	0	0	0	0	4
	Tier Totals	5	2	4	4	4	2	4	5	3	4	3	40

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	Point Total
	5	4	3	5	17

Note:
1. Ensure that at least two topics from every K/A category are sampled within each tier (i.e., the "Tier Totals" in each K/A category shall not be less than two).
2. Actual point totals must match those specified in the table.
3. Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.
4. Systems/evolutions within each group are identified on the associated outline.
5. The shaded areas are not applicable to the category/tier.
6.* The generic K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.
7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings for the RO license level, and the point totals for each system and category. K/As below 2.5 should be justified on the basis of plant-specific priorities. Enter the tier totals for each category in the table above.

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295003 Partial or Complete Loss of AC Pwr / 6	X		X				AK1.04 - 3.2; AK3.06 - 3.7		2
295006 SCRAM / 1						X	G2.1.19 - 3.0		1
295007 High Reactor Pressure / 3			X				AK3.04 - 4.1		1
295009 Low Reactor Water Level / 2	X		X				AK1.01 - 2.9; AK3.01 - 3.3		2
295010 High Drywell Pressure / 5						X	G2.4.6 - 4.0		1
295013 High Suppression Pool Temp. / 5						X	G2.2.22 - 4.1		1
295014 Inadvertent Reactivity Addition / 1				X	X		AA1.06 - 3.4; AA2.01 - 4.2		2
295015 Incomplete SCRAM / 1		X					AK2.02 - 3.7		1
295016 Control Room Abandonment / 7				X			AA1.07 - 4.3 (No K/A's in K1)		1
295017 High Off-site Release Rate / 9				X X			AA1.03 - 3.4; AA1.12 - 3.9		2
295023 Refueling Accidents Cooling Mode / 8		X					AK2.03 - 3.6		1
295024 High Drywell Pressure / 5		X					EK2.04 - 3.9		1
295025 High Reactor Pressure / 3		X			X		EK2.04 - 4.1; EA2.04 - 3.9		2
295026 Suppression Pool High Water Temp. / 5			X				EK3.05 - 4.1		1
295027 High Containment Temperature / 5							(MK III Containment only)		-
295030 Low Suppression Pool Water Level / 5	X					X	EK1.02 - 3.8; G2.4.18 - 3.6		2
295031 Reactor Low Water Level / 2				X			EA1.03 - 4.4		1
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	X				X		EK1.04 - 3.6; EA2.01 - 4.3		2
295038 High Off-site Release Rate / 9					X		EA2.03 - 4.3		1
500000 High Containment Hydrogen Conc. / 5						X	G2.4.22 - 4.0		1
K/A Category Totals:	4	4	4	5	4	5	Group Point Total:		26

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4		x					AK2.07 - 3.4		1
295002 Loss of Main Condenser Vacuum / 3		X					AK2.06 - 2.7		1
295004 Partial or Total Loss of DC Pwr / 6				X			AA1.02 - 4.1		1
295005 Main Turbine Generator Trip / 3							(Not selected)		-
295008 High Reactor Water Level / 2				X			AA1.03 - 3.1		1
295011 High Containment Temperature / 5							(MK III Containment only)		
295012 High Drywell Temperature / 5					X		AA2.02 - 4.1		1
295018 Partial or Total Loss of CCW / 8			X				AK3.02 - 3.4		1
295019 Partial or Total Loss of Inst. Air / 8			X				AK3.01 - 3.2		1
295020 Inadvertent Cont. Isolation / 5 & 7							(Not selected)		-
295021 Loss of Shutdown Cooling / 4	X						AK1.04 - 3.7		1
295022 Loss of CRD Pumps / 1		X					AK2.03 - 3.4		1
295028 High Drywell Temperature / 5					X		EA2.01 - 4.1		1
295029 High Suppression Pool Water Level / 5			X				EK3.01 - 3.9		1
295032 High Secondary Containment Area Temperature / 5		X					EK2.01 - 3.6		1
295033 High Secondary Containment Area Radiation Levels / 9						X	G2.4.6 - 4.0		1
295034 Secondary Containment Ventilation High Radiation / 9						X	G2.1.20 - 4.2		1
295035 Secondary Containment High Differential Pressure / 5			X				EK3.02 - 3.5		1
295036 Secondary Containment High Sump/Area Water Level / 5						X	G2.4.24 - 3.7		1
600000 Plant Fire On Site / 8				X			AA1.08 - 2.9		1
K/A Category Point Totals:	1	4	4	3	2	3	Group Point Total:		17

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201005 RCIS												(BWR-6 only)		-
202002 Recirculation Flow Control				X								K4.06 - 3.1		1
203000 RHR/LPCI: Injection Mode	X											K1.07 - 3.3		1
206000 HPCI				X						X		K4.17 - 3.4; A4.10 - 3.5		2
207000 Isolation (Emergency) Condenser												(BWR-2,3 only)		-
209001 LPCS											X	G2.2.18 - 3.6		1
209002 HPCS												(BWR-5,6 only)		-
211000 SLC											X	G2.2.23		1
212000 RPS			X									K3.06 - 4.1		1
215004 Source Range Monitor					X							K5.03 - 2.8		1
215005 APRM / LPRM			X									K3.01 - 4.0		1
216000 Nuclear Boiler Instrumentation							X					A1.01 - 3.3		1
217000 RCIC								X				A2.10 - 3.1		1
218000 ADS					X							K5.01 - 3.8		1
223001 Primary CTMT and Auxiliaries						X						K6.09 - 3.6		1
223002 PCIS/Nuclear Steam Supply Shutoff	X											K1.12 - 3.3 (K5-No K/A's)		1
226001 RHR/LPCI: CTMT Spray Mode											X	G2.4.17 - 3.8		1
239002 SRVs								X				A2.03 - 4.2		1
241000 Reactor/Turbine Pressure Regulator								X				A2.04 - 3.8 (K2-All K/A's < 2.5 for SROs)		1
259002 Reactor Water Level Control							X					A1.01 - 3.8		1
261000 SGTS		X										K2.03 - 2.5		1
262001 AC Electrical Distribution									X			A3.02 - 3.3		1
264000 EDGs				X						X		K4.05 - 3.5; A4.03 - 3.4		2
290001 Secondary CTMT	X											K1.08 - 3.3 (K2-No K/A's)		1
K/A Category Point Totals:	3	1	2	3	2	1	2	3	1	2	3	Group Point Total:		23

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201001 CRD Hydraulic									X			A3.08 - 2.9		1
201002 RMCS	X											K1.05 - 3.6 (K2-All K/A's < 2.5 for SROs; K5-No K/A's)		1
201004 RSCS												(N/A for Hatch)		-
201006 RWM												(Not selected)		-
202001 Recirculation								X				A2.01 - 3.9		1
204000 RWCU												(Not selected) (K2-All K/A's < 2.5 for SROs)		-
205000 Shutdown Cooling			X									K3.01 - 3.3		1
214000 RPIS												(Not selected) (K2, A1-No K/A's)		-
215002 RBM								X				A2.03 - 3.3		1
215003 IRM												(Not selected)		-
219000 RHR/LPCI: Torus/Pool Cooling Mode												(Not selected)		-
230000 RHR/LPCI: Torus/Pool Spray Mode										X		A4.02 - 3.6		1
234000 Fuel Handling Equipment					X							K5.02 - 3.7 (K2-No K/A's)		1
239003 MSIV Leakage Control												(N/A for Hatch)		-
245000 Main Turbine Gen. and Auxiliaries												(Not selected)		-
259001 Reactor Feedwater										X		A4.01 - 3.5		1
262002 UPS (AC/DC)												(Not selected) (K2-No K/A's)		-
263000 DC Electrical Distribution			X									K3.03 - 3.8		1
271000 Offgas												(Not selected) (K2-All K/A's < 2.5 for SROs)		-
272000 Radiation Monitoring									X			A3.02 - 3.7		1
286000 Fire Protection					X							K5.05 - 3.1		1
290003 Control Room HVAC												(Not selected) (K2-All K/A's < 2.5 for SROs)		-
300000 Instrument Air		X										K2.01 - 2.8 (A1-No K/A's)		1
400000 Component Cooling Water							X					A1.04 - 2.8 (K5-All K/A's < 2.5 for SROs)		1
K/A Category Point Totals:	1	1	2	0	2	0	1	2	2	2	0	Group Point Total:		13

Plant Systems - Tier 2/Group 3

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201003 Control Rod and Drive Mechanism												(Not selected) (K2-No K/A's)		-
215001 Traversing In-core Probe						X						K6.04-3.4 (K2-All K/A's < 2.5 for SROs)		1
233000 Fuel Pool Cooling and Cleanup				X								K4.06 - 3.5		1
239001 Main and Reheat Steam	X											K1.27 - 4.1		1
256000 Reactor Condensate												(Not selected)		-
268000 Radwaste							X					A1.01 - 3.1 (K2,K4,A3-No K/A's)		1
288000 Plant Ventilation												(Not selected) (K2,A1-All K/A's < 2.5 for SROs)		
290002 Reactor Vessel Internals												(Not selected) (K2,A1,A3,A4-No K/A's)		
K/A Category Point Totals:	1	0	0	1	0	1	1	0	0	0	0	Group Point Total:		4

Plant-Specific Priorities

System / Topic	Recommended Replacement for...	Reason	Points
None			
Plant-Specific Priority Total (limit 10):			

Facility: Hatch	Date of Exam: 10/29/99		Exam Level: SRO	
Category	**K/A #**	**Topic**	**Imp.**	**Points**
Conduct of Operations	2.1.3	Knowledge of shift turnover practices	3.4	1
	2.1.4	Knowledge of shift staffing requirements	3.4	1
	2.1.19	Knowledge of conditions & limitations in the facility license	3.9	1
	2.1.22	Ability to determine Mode of Operation	3.3	1
	2.1.34	Ability to maintain primary & secondary plant chemistry within allowable limits	2.9	1
	Total			
Equipment Control	2.2.6	Knowledge of the process for making changes in procedures as described in FSAR	3.3	1
	2.2.14	Knowledge of the process for making configuration changes	3.0	1
	2.2.21	Knowledge of pre and post maintenance operability requirements	3.5	1
	2.2.32	Knowledge of the effects of alterations on core configuration	3.3	1
	Total			
Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements	3.0	1
	2.3.4	Knowledge of radiation exposure limits and contamination control including permissible levels in excess of those authorized	3.1	1
	2.3.9	Knowledge of the process for performing a containment purge	3.4	1
	Total			
Emergency Procedures/ Plan	2.4.18	Knowledge of the specific bases for EOPs	3.6	1
	2.4.19	Knowledge of EOP layout, symbols, and icons	3.7	1
	2.4.37	Knowledge of the lines of authority during an emergency	3.5	1
	2.4.38	Ability to take actions called for in the facility emergency plan including (if required) supporting or acting as emergency coordinator	4.0	1
	2.4.44	Knowledge of emergency plan PARs	4.0	1
	Total			
Tier 3 Point Total (RO/SRO)				17

Facility: Hatch (99-301) Date of Exam: 10/29/99 Exam Level: ~~RO~~/SRO

	Item Description				Initial a	Initial b*	Initial c#
1.	Questions and answers technically accurate and applicable to facility				DB	N/A	ME
2.	a. NRC K/As referenced for all questions b. Facility learning objectives referenced as available				DB	N/A	ME
3.	RO/SRO overlap is no more than 75 percent, and SRO questions are appropriate per Section D.2.d of ES-401				DB	N/A	ME
4.	No more than 25 questions are duplicated from [practice exams, quizzes, and] the last two NRC licensing exams; enter the actual number of duplicated questions at right		NRC: 18	Other: 0	DB	N/A	ME
5.	[No (Less than 5 percent) question duplication from the license screening/audit exam (if independently written)]				N/A	N/A	N/A
6.	Bank use meets limits (no more than 50 percent from the bank, at least 10 percent new, and the rest modified); enter the actual question distribution at right	Bank: 43	Modified: 12	New: 45	DB	N/A	ME
7.	Between 50 and 60 percent of the questions on the exam (including 10 new questions) are written at the comprehension/analysis level; enter the actual question distribution at right	Memory: 43		C/A: 57	DB	N/A	ME
8.	References/handouts provided do not give away answers				DB	N/A	ME
9.	Question distribution meets previously approved examination outline; deviations are justified				DB	N/A	ME
10.	Question psychometric quality and format meet ES, Appendix B, guidelines				DB	N/A	ME
11.	The exam contains 100, one-point, multiple choice items; the total is correct and agrees with value on cover sheet				DB	N/A	ME

	Printed Name / Signature	Date
a. Author	D.C. PAYNE / D.C. Payne	10/26/99
b. Facility Reviewer(*)	N/A	N/A
c. NRC Chief Examiner(*)	ME ERNSTES / Mike Ernstes	10/27/99
d. NRC Regional Supervisor(*)	HDChristensen / HDChristensen	10/27/99

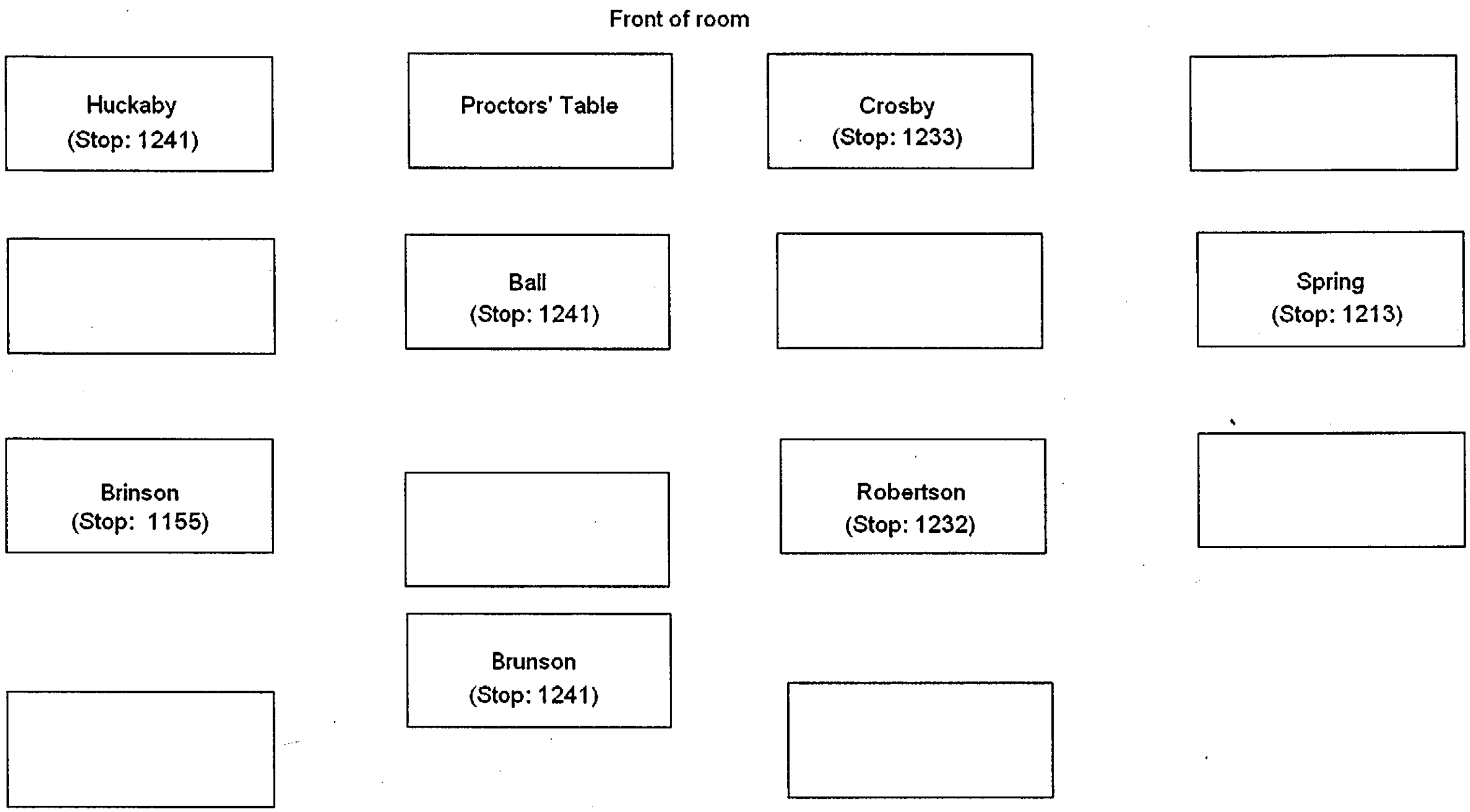
Note: * The facility reviewer's signature is not applicable for NRC-developed examinations; two independent NRC reviews are required.
See special instructions (Section E.2.c) for Items 1, 4, 5, and 6.
[ ] The items in brackets do not apply to NRC-prepared examinations.

HATCH SRO WRITTEN EXAM
10/29/99

Candidate Questions and Answers

Candidate's Name	Question	Answer
Robertson	Q#47: In the initial conditions, does STOP mean P-T-L?	Ans: The switch is actually pointing to the STOP position. Whatever it takes for the switch to stay in that position, has occurred.
Crosby	Q#47: Asked the same question as Robertson above.	Ans: Provided same answer as above. Also, provided clarification to rest of candidates.
Ball	Q#93: For distractor "a", I'm assuming the 'sample line' is a primary system, right?	Ans: It is a line to the reactor sample sink and comes from the reactor vessel.
Ball	Q#24: The question talks about the "next step" meaning proceeding to the core and getting another assembly. So you are looking for which one of the answer options will cause a rod block during that process, right?	Ans: Yes. Which one of the answer options causes a rod block during the process of going from the fuel pool to the reactor, getting an assembly and returning to the pool.
Brunson	Q#91: In initial conditions, the statement "No refuel floor airspace openings exist on either unit" means there is no communication between the refuel floor and either unit, right?	Ans: Yes. No air communication exists between the refuel floor and either unit.
Robertson	Q#14: For distractor "c", does 'protective actions' include evacuation? On site right?	Ans: It includes everything that might be evaluated or implemented to protect personnel (on site).

NOTE: The light above Mr. Ball's desk turned off and on about 10 times during the course of the exam.



Start Time: 0741

Stop Time: 1241

Hatch SRO Written Examination
10/29/99

Seating Chart

DOOR

Facility: Hatch (99-301) Date of Exam: 10/29/99 Exam Level: ~~RO~~/**SRO**

	Item Description	Initials a	Initials b	Initials c
1.	Answer key changes and question deletions justified and documented	n/a	N/A	
2.	Applicants' scores checked for addition errors (reviewers spot check > 25% of examinations)		N/A	
3.	Grading for all borderline cases (80% +/- 2%) reviewed in detail	n/a	N/A	N/A
4.	All other failing examinations checked to ensure that grades are justified	n/a	N/A	N/A
5.	Performance on missed questions checked for training deficiencies and wording problems; evaluate validity of questions missed by half or more of the applicants		N/A	

	Printed Name / Signature	Date
a. Grader	Ronald F. Aiello	11/10/99
b. Facility Reviewer(*)	N/A	N/A
c. NRC Chief Examiner (*)	D. Charles Payne	11/22/99
d. NRC Supervisor (*)	Harold O. Christensen	11/26/99

(*) The facility reviewer's signature is not applicable for examinations graded by the NRC; two independent NRC reviews are required.