March 17, 1999

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

FROM: Virgil Curley, Licensing Assistant
Operating Licensing Branch, R I

SUBJECT: OPERATOR LICENSING EXAMINATION ADMINISTERED ON Sept. 11-22, 1998, AT Peach Bottom.
DOCKET #50-277 and 50-278

On ________________ 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.

Please return copy of this package
To Region I ATTN: V. Curley
Thanks

DRS Master
EXAM File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

October 30, 1998

Mr. G. Rainey
President
PECO Nuclear
Nuclear Group Headquarters
Correspondence Control Desk
P. O. Box 195
Wayne, Pennsylvania 19087-0195

SUBJECT: EXAMINATION REPORT NOS. 50-277/98-301 AND 278/98-301 (OL)

Dear Mr. Rainey:

During the period of September 14, 1998 to September 22, 1998, the facility and NRC administered initial examinations to 12 employees of your company who had applied for licenses to operate the Peach Bottom Atomic Power Station, Units 2 and 3. The examination was prepared by your staff using the Examination Standards, NUREG 1021, Interim Revision 8, as guidance. All applicants passed all portions of the examinations and received licenses. The applicants were well prepared for the examinations..

The initial submittal of the written examination was unacceptable. We determined that 28% of the written examination questions were inadequate for several reasons as noted herein. However, your staff now appears to understand our expectations as embodied in the Examination Standards. Your cooperation with us in this important matter was appreciated.

Should you have any questions regarding this information, please contact Carl Sisco of my staff at (610) 337-5076 or me at (610) 337-5183.

Sincerely,

Richard J. Conte

Richard J. Conte, Chief
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

Docket Nos. 50-277; 50-278

IE42

~~9811060101~~ 3pp

cc w/encl; w/Attachments 1-4:
L. MacEntee, Acting Director - Training

cc w/encl; w/o Attachments 1-4:
J. Hagan, Vice President, Nuclear Station Support
J. Doering, Vice President, Peach Bottom Atomic Power Station
M. Warner, Plant Manager, Peach Bottom Atomic Power Station
G. D. Edwards, Chairman, Nuclear Review Board
R. Boyce, Director, Nuclear Quality Assurance
A. F. Kirby, III, External Operations - Delmarva Power & Light Co.
G. J. Lengyel, Manager, Experience Assessment
J. W. Durham, Sr., Senior Vice President and General Counsel
T. M. Messick, Manager, Joint Generation, Atlantic Electric
W. T. Henrick, Manager, External Affairs, Public Service Electric & Gas
R. McLean, Power Plant Siting, Nuclear Evaluations
D. Levin, Acting Secretary of Harford County Council
R. Ochs, Maryland Safe Energy Coalition
J. H. Walter, Chief Engineer, Public Service Commission of Maryland
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance
Mr. & Mrs. Kip Adams
Commonwealth of Pennsylvania
State of Maryland
TMI - Alert (TMIA)

<u>Distribution w/encl; w/Attachments 1-4:</u>
DRS Master Exam File
PUBLIC
Nuclear Safety Information Center (NSIC)

<u>Distribution w/encl; w/o Attachments 1-4</u>:
Region I Docket Room (with concurrences)
J. Wiggins, DRS
L. Nicholson, DRS
C. Sisco, Chief Examiner, DRS
C. Anderson, DRP
D. Florek, DRP
R. Junod, DRP
NRC Resident Inspector
DRS OL Facility File
DRS File

<u>Distribution w/encl; w/o Attachments 1-4</u>: **VIA E-MAIL**
B. McCabe, OEDO
R. Capra, PDI-2, NRR
Inspection Program Branch, NRR (IPAS)
M. Thadani, NRR
B. Buckley, NRR
R. Correia, NRR
M. Campion, ORA
DOCDESK

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket Nos.: 50-277; 50-278

Report Nos: 98-301 (OL)

Licensee: PECO Nuclear
Wayne, PA

Facility: Peach Bottom Unit Nos. 2 and 3

Location: Delta, PA

Exam Date: September 14-22, 1998

Examiners: John Caruso, Operations Engineer
Steven Dennis, Operations Engineer

Chief Examiner: Carl Sisco, Operations Engineer, Operator Licensing and
Human Performance Branch
Division of Reactor Safety

Approved By: Richard J. Conte, Chief, Operator Licensing and
Human Performance Branch
Division of Reactor Safety

~~9810060137~~ MY

EXAMINATION SUMMARY

Peach Bottom Atomic Power Station
Examination Report 50-277/98-301 and 50-278/98-301 (OL)

Operations

Two senior reactor operator upgrade, six instant and four reactor operator applicants were administered initial licensing exams. All applicants passed the examination, were well prepared, and received licenses.

The initial submittal of the written examination was identified by NRC staff to be unacceptable. The examination contained multiple examples of questions that were inadequate for several reasons as noted herein. PECO staff subsequently revised the written examination in a timely manner.

The facility provided six post examination comments that were resolved but were of a technical nature indicating that a more detailed review of the reference material used in the construction of the examination was needed.

Report Details

I. Operations

05 Operator Training and Qualifications

05.1 Reactor Operator and Senior Reactor Operator Initial Exams

a. Scope

The Facility and NRC administered initial examinations to six SRO instants, two upgrade and four RO applicants. The examinations were prepared and administered in accordance with NUREG-1021, "Examiner Standards," (ES) interim Revision 8. Personnel involved in the development of the examination signed security agreements to assure integrity of the examination process. The written examination was administered by the PECO staff and the NRC administered the operating portion.

b. Observations and Findings

Examination Grading and Overall Results

The results of the examinations are summarized below:

	SRO Pass/Fail	RO Pass/Fail
Written	8/0	4/0
Operating	8/0	4/0
Overall	8/0	4/0

Examination Preparation

The examination sample plan and the proposed examination was prepared by PECO staff and was received in the NRC regional office in a timely manner. The sample plan was comprehensive and needed no changes as a result of the in-office review. The proposed examination was very well organized and added to the ease of review. The on-site review of the proposed operating test scenarios indicated they were well defined and required no changes. The Job Performance Measures (JPMs) and supporting questions as well as the administrative portion of the examination required minor administrative enhancements to be made.

The initial submittal of the written examination was found to be unacceptable in that approximately 28% of the questions were deemed to be inadequate test items by the NRC staff. An in-office meeting was conducted with PECO staff members to discuss the written examination on August 25, 1998. The written examination questions were unacceptable for several different reasons in that certain questions had no right answer or multiple correct answers and answers were technically incorrect or failed to discriminate at the appropriate license level. In addition, questions that attempted to test more than one concept resulted in no correct answer. PECO staff was in agreement with the deficient questions and committed to revise the examination. The NRC conducted an onsite review of the examination the week of September 8, 1998. During the onsite review, the NRC reviewed the changes to the written examination made by PECO staff. The review indicated that the changes made to the written examination in response to the NRC identified deficiencies were acceptable.

Attachments 1 and 2 address facility comments and the NRC resolution of these comments on the written examinations. All comments were of a technical nature for which the facility was responsible prior to examination administration. The results of comment resolution were three questions deleted and three questions had two correct answers. In accordance with the ES, these changes did not invalidate the examination. However, these technical problems could have been prevented during examination preparation by a more detailed review of the reference material used in the construction of the examination.

Operating Examination

Two of four examination crews failed to attempt to isolate a primary system discharging into the secondary containment during the dynamic scenarios. The indications were of a leak from the HPCI system steam line and the scenario was designed for the leak to be unisolable. However, had the scenario been designed for the success path to be for the applicants to isolate the HPCI steam line, their failure to do so may have allowed the plant to unnecessarily be in a degraded condition. This information is provided for feedback to the training program.

No simulator fidelity problems were identified.

c. Conclusions

Two senior reactor operator upgrade, six instant and four reactor operator applicants were administered initial licensing exams. All applicants were well prepared, and received licenses. The applicants passed the examination.

The initial submittal of the written examination was identified by NRC staff to be unacceptable. The examination contained multiple examples of questions that were inadequate for several reasons as noted herein. PECO staff subsequently revised the written examination in a timely manner.

The facility provided six post examination comments that were resolved but were of a technical nature indicating that a more detailed review of the reference material used in the construction of the examination was needed.

O5.2 Licensed Operator Eligibility

The inspector determined that UFSAR section 13.2 was revised and questioned if the licensee made an unreviewed safety question determination in accordance with 10 CFR 50.59. Specifically, wording of "....staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, and subsequent revisions of this standard (i.e. ANSI/ANS 3.1-1981 and ANSI/ANS 3.1-1987), as appropriate, ..." was included in earlier revisions of the UFSAR and the italic words were removed from the UFSAR Revision 5 (4/98). During subsequent discussions with PECO staff and the review of ECR 98-00055, the inspector determined the UFSAR revision was an administrative change to document a change in commitment between PECO and the state of Pennsylvania. This agreement was reached in 1991 and expired in 1997. The need to perform a 50.59 review was reviewed by PECO and determined that a 50.59 review was not necessary. The italic words in the UFSAR were in effect a "place keeper" for PECO staff to recognize a commitment made to another organization. The inspector concurred that a 50.59 review was not necessary in this situation.

M8 Miscellaneous Maintenance Issues

CLOSED Violation (50-277;278/98-004-05): failure to include the area radiation monitoring system within the scope of the maintenance rule in accordance with 10 CFR 50.65a(1). The inspector reviewed the scoping documentation and determined the system in within the scope of the maintenance rule. Licensee corrective actions included an extent of conditions review.

V. Management Meetings

X1 Exit Meeting

A conference call was conducted with PECO staff on October 5, 1998 to present the examination results. The unacceptable submittal of the written examination was discussed. PECO staff agreed with the inspection findings and stated they would endeavor to improve the written examination submittals.

ITEMS OPEN, CLOSED AND DISCUSSED

Open

None.

Closed

50-277/98-004-05	VIO	Failure to include the area radiation monitoring system within
50-278/98-004-05	VIO	maintenance rule.

Attachment 1: Resolution of Facility Comments
Attachment 2: Facility Comments on Written Examination
Attachment 3: Senior Reactor Operator and Reactor Operator Written Examination

ATTACHMENT 1

Resolutions of Facility Comments

(SRO #2/RO #2)- Plant procedure OM-C-6.2 Temporary Relief was revised after examination validation. The revision made all answers to the question incorrect. PECO recommended, and the NRC agreed that the question be deleted from the examination.

(SRO #22/RO #20)- The question had no correct answer because, for the recirculation pump motor-generator set, it is the generator over current and not the motor feeder over current that results in the equipment trip. PECO recommended, and the NRC agreed that the question be deleted from the examination.

(SRO #23/RO #21)- The question had no correct answer because the oil temperature actually decreases and not increases on a recirculation pump speed increase. PECO recommended, and the NRC agreed that the question be deleted from the examination.

(SRO #50/RO #55)- The question had two correct answers because the core spray pump will start automatically in response to the conditions stated in the question. PECO recommended, and the NRC agreed that answers "a" and "b" are both correct.

(SRO #54/RO #59)- The question had two correct answers because the cause of the Fast Transfer logic for a 4Kv electrical bus not stated. PECO recommended, and the NRC agreed that answers "a" and "c" are both correct.

(SRO #55/RO #62)- The question had two correct answers because the inboard MSIVs will close on area high temperatures. PECO recommended, and the NRC agreed that answers "b" and "c" were both correct.

ATTACHMENT 2

Facility Comments on Written Examination

John Doering, Jr.
Vice President
Peach Bottom Atomic Power Station

PECO NUCLEAR
A Unit of PECO Energy

PECO Energy Company
1848 Lay Road
Delta, PA 17314-9032
717 456 4000
Fax 717 456 4243
E-mail: jdoering@peco-energy.com

September 25, 1998

Mr. Richard J. Conte
Chief Operator, Licensing - Human Performance Branch
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Dear Mr. Conte:

On September 11, 1998, the NRC Written Examination was administered at Peach Bottom Atomic Power Station. In accordance with ES-402, a thorough review of this exam has been completed with the license candidates and station technical staff. This review has resulted in comments on Master Exam Question numbers 2, 30, 31, 69, 74, and 77. Each of the attached comments includes a copy of the question, recommended action, justification, and references.

Please contact Phil Nielsen at (717) 456-7014, ext. 3497 for any additional information or clarification.

Sincerely,

J Doering

Vice President, PBAPS

JD/JMA/PEN:clg

Enclosures (6)

bcc:	J. A. Bernstein	63C-7, Chesterbrook
	R. L. Gambone	A4-1S, Peach Bottom
	L. E. MacEntee	PB-TC, Peach Bottom
	A. N. Tarbert	A4-5S, Peach Bottom
	M. E. Warner	A4-1S, Peach Bottom
	Nuclear Records	Peach Bottom

Peach Bottom Atomic Power Station
Written NRC Examination Facility Comments
September 1998

Master Question #2 (SRO #2 / RO #2)

Recommended Action: Delete

Justification:

The question was based on a procedure which was revised between the time of exam validation/approval and exam administration. The candidates (and all of the Operations Department) were notified of the OM change via vcice mail.

OM-C-6.2, Temporary Relief, Revision 2 (dated 9/4/98), step 2.5 now states that while attending the shift turnover meeting is preferred, it is not required. This change makes answer "A" incorrect since it states that the individual "shall have attended the shift turnover meeting". The new procedure makes all of the potential answers to this question incorrect.

Attached References:

Master Question #2

OM-C-6.2, Temporary Relief, Revision 1

OM-C-6.2, Temporary Relief, Revision 2

Peach Bottom Atomic Power Station
Written NRC Examination Facility Comments
September 1998

Master Question #30 (SRO #22 / RO #20)

Recommended Action: Delete

Justification:

This question was designed to evaluate the operators knowledge of what would cause a trip of the Recirculation Pump Drive Motor Breaker if the required operator actions were not taken on a pump start with a scoop tube lock. The question was based on a step in the Recirculation Pump startup procedure. Research by the System Manager has shown that the words in the procedure did not precisely state which overcurrent trip would occur. The answer to the examination question indicated that the breaker would trip on "motor *feeder* overcurrent". In fact, the trip would be caused by *generator* overcurrent. This trip is normally bypassed for ten seconds on a Recirculation Pump start. This is shown on the attached and highlighted prints.

Given this information, there is no correct answer.

The indicated procedural reference, SO 2A.1.A-2, "Starting the First Recirculation Pump", has been revised to clarify the cause of the trip. A Procedure Change has been initiated for the associated Unit 3 procedure.

Attached References:

Master Question # 30

E-171 Sheet 3

M-1-S-2 Sheets 5 and 7

SO 2A.1.A-2, Starting the First Recirculation Pump, Revision 17

SO 2A.1.A-2, Starting the First Recirculation Pump, Revision 18
(NOTE: This revision is approved and on hold until system modification completion in mid-October.)

Peach Bottom Atomic Power Station
Written NRC Examination Facility Comments
September 1998

Master Question #31 (SRO #23 / RO #21)

Recommended Action: Delete

Justification:

This question was written with the expectation that as recirculation pump speeds were increased in support of a power ascension, the lube oil temperatures would go up. In fact, a discussion with the system manager has revealed that as recirculation pump speeds are increased for this type of power ascension, lube oil temperatures will actually drop. This was demonstrated by a situation at Peach Bottom recently when high system temperatures were experienced during a power reduction and returned to normal as recirculation speeds were increased. The attached Action Requests (A1166231, A1166232, and A1099378) document this occurrence.

Given this information, there is no correct answer.

Attached References:

Master Question # 31

Action Request A1166231 - B MG Lube Oil Output End Bearing Temperature

Action Request A1166232 - A MG Lube Oil Output End Bearing Temperature

Action Request A1099378 - Unit 2 – Hi Recirc M/G Set Lube Oil Temperatures

Peach Bottom Atomic Power Station
Written NRC Examination Facility Comments
September 1998

Master Question #69 (SRO #50 / RO #55)

Recommended Action: Accept both answers "b" and "d"

Justification:

This question assumes that a Core Spray Pump is running for a surveillance test. The question evaluates the response of the Core Spray Pump logic to an undervoltage fast transfer condition.

When a low voltage condition occurs on the bus, a relay (127X-15, E-183 Sheet 1) in the Core Spray Pump logic will cause the pump to trip. This undervoltage trip signal clears as soon as power is restored to the bus. The coils and relays for this sequence are highlighted on the attached prints (E-193 Sheet 1 and E-47 Sheet 2).

The original answer (d) to the question acknowledged the fact that the pump would not restart automatically under these conditions and required manual operator action to restart. This answer is correct.

Answer "b" changed the conditions to include a Core Spray system initiation signal. A review of the print shows that the initiation logic closes a relay (14A-K12A, E183 Sheet 1) in parallel with the hand control switch. With no trip signal existing, this automatic start signal would result in a start of the pump. This answer is also correct.

Attached References:

Master Question # 69

E-183 Sheet 1

E-193 Sheet 1

E-47 Sheet 2

Peach Bottom Atomic Power Station
Written NRC Examination Facility Comments
September 1998

Master Question #74 (SRO #54 / RO #59)

Recommended Action: Accept both answers a and c

Justification:

With the diesel generator control switch in Pull-to-Lock, the diesel will still start when it receives a 2 psig drywell pressure signal.

The question indicates that the 4KV bus has experienced a failure to Fast Transfer. Whether or not the output breaker closes depends on what caused the failure to Fast Transfer. The question does not specify the cause of the failure and during the administration of the examination, this was questioned. The proctor did not provide any additional information.

If the failure to Fast Transfer was related to a failure of the other off-site breaker, then the diesel generator output breaker would be expected to close. This makes answer "c" correct.

If the failure to Fast Transfer was due to a failure of the Fast Transfer logic, then the diesel generator output breaker would not close. This makes answer "a" correct.

The diesel generator breaker closure logic can be viewed on the attached and highlighted E-193 Sheet 3. In order for the output breaker to automatically close, contacts from the Fast Transfer logic, 102-1701 or 102-1708, must close.

Answers "a" or "c" are both correct depending on the cause of the failure to Fast Transfer.

Attached References:

Master Question # 74

E-193 Sheet 3

E-71 Sheet 5

Peach Bottom Atomic Power Station
Written NRC Examination Facility Comments
September 1998

Master Question #77 (SRO #55 / RO #62)

Recommended Action: Accept both answers "b" and "c"

Justification:

The ON-119, Loss of Instrument Air, Bases Step 2.2 (attached) indicates that "at an Instrument Air header pressure of 75 psig, Reactor Building Dampers will have failed closed. The loss of ventilation will result in elevated main steam line tunnel temperatures and resultant Group I isolation."

Answer "b" states that "The Reactor Building Ventilation system will no longer support normal building heat loads". This is correct and was the expected response.

Answer "c" states that "The Inboard Main Steam Isolation Valves will fast close (3 to 5 seconds)". According to the bases above when the Reactor Building Dampers fail closed, a Group I isolation will occur. When a High Steam Tunnel Temperature Group I isolation occurs, the Main Steam Isolation Valves will fast close. This makes answer "c" also a correct answer.

Answers "b" and "c" are both correct.

Attached References:

Master Question # 77

ON-119, Loss of Instrument Air Bases, Step 2.2

ATTACHMENT 3

SRO/RO Written Examinations

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

Applicant Information

Name:	Region: I
Date: 09/11/98	Facility: Peach Bottom
License Level: RO	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 four 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

PART 1

KEY

PECO NUCLEAR

STATION PB / LGS

COURSE TITLE LOT 9701 NRC EXAM FORM RO

ANSWER KEY

NAME PRINT ______ last ______ first ______ mi

IMPORTANT

- USE #2 PENCIL
- EXAMPLE: ‹A› ‹B› ‹C› ‹D› ‹E›
- ERASE *COMPLETELY* TO CHANGE

SOCIAL SECURITY NUMBER ______

COMPANY / PECO PAYROLL # ______

DATE ______

I HAVE REVIEWED AND UNDERSTAND THE CORRECTED QUIZ; ALL WORK ON THIS EXAMINATION IS MY OWN, I HAVE NEITHER GIVEN NOR RECEIVED ASSISTANCE ______ signature

DELETED

DELETED

DELETED

50

50

1 Given the following conditions:

- A Reactor Operator (RO) has just returned to shift following 2 weeks vacation
- The RO's license is active and current
- On the first day back on shift, this RO worked a normal 12-hour shift and then accepted and worked 2 hours of overtime to cover for an absent operator

What is the maximum number of hours this RO may work on the next day (his second day back on shift)? Assume no additional authorizations have been made.

a. 8 hours

b. 10 hours

c. 12 hours

d. 14 hours

2 The Unit 2 Unit Reactor Operator (URO) is being temporary relieved from his duties at the control boards for a break.

The individual providing the temporary relief shall:

a. walkdown all Unit 2 control panels and shall have attended the Shift Turnover Meeting.

b. perform a complete turnover in accordance with OM-C-6.1, "Shift Turnover" and shall NOT hold the URO position for more than 60 minutes.

c. perform a complete turnover in accordance with OM-C-6.1, "Shift Turnover", and have attended the Shift Turnover Meeting.

d. walkdown all Unit 2 control panels and shall NOT hold the URO position for more than 30 minutes.

3 Given the following conditions:

- Unit 3 is performing a reactor and plant startup in accordance with GP-2, "Normal Plant Startup"
- All control rods are fully inserted
- Reactor coolant temperature is 185 °F

Assuming all requirements are met, the Unit will be in Mode 2 when:

a. reactor temperature exceeds 212 °F.

b. the first control rod is withdrawn.

c. the reactor is critical.

d. the Mode Selector Switch is placed in "Startup/Hot Standby".

4 An operator, performing an Independent Verification of a check-off list (COL), discovers that a manually operated valve is danger tagged in the "open" position? The COL required position for the valve is "closed". Which of the following describes the required action(s)?

a. The COL step should NOT be initialed, the clearance number and valve position should be noted on the COL.

b. The COL position should be changed to the actual valve position, then the step should be initialed and dated.

c. The COL step should be marked "N/A" and the remainder of the COL should be completed.

d. The COL should NOT be completed until a temporary change noting the discrepancy is prepared in accordance with A-3.

5 Given the following conditions:

- Unit 3 is performing a shutdown with control rod insertions in progress
- Reactor power is 16% with generator output at 180 MWe
- The Plant Reactor Operator is deinerting the drywell
- The CRS is monitoring the shutdown from the CRS desk

Which of the following additional requirements, if met, would allow a License Class Instant SRO trainee under direction of the shutdown Reactor Operator to continue rod motion for the given conditions?

a. The Reactor Engineer is present to satisfy Technical Specification requirements.

b. The 4th Reactor Operator is performing Double Verification for control rod movements.

c. The Shift Manager has conducted a shift briefing on control rod insertions.

d. The Vice President-PBAPS written permission has been received allowing trainees to manipulate control rods.

6 Given the following conditions:

- A 480 VAC MCC breaker is danger tagged "open"
- Sometime later, the breaker is being removed from its compartment for maintenance

Which of the following describes the required tagging action for the given conditions?

The danger tag shall:

a. be removed from the breaker handle and applied to the breaker compartment door.

b. remain on the breaker handle and a suspension label installed on the breaker compartment door.

c. remain on the breaker handle, the breaker removed from the cubicle and an additional danger tag installed on the breaker compartment door..

d. be removed from the breaker handle but kept active and maintained in the physical possession of Operations while the breaker is out of the compartment.

7 Which of the following would REQUIRE a second individual to actually witness the activity while it is occurring?

a. Restoration of a throttled valve to its required locked position.

b. Fuse removal as directed by the T-200 procedures.

c. Restoration of a clearance on a ECCS System.

d. A routine surveillance test being performed in a Radiation Area.

8 Given the following conditions:

- A male, fully qualified Peach Bottom radiation worker is scheduled to work in a Level I - Locked High Radiation Area (Assume area radiation levels are at the minimum for the area classification)
- Total Effective Dose Equivalent TEDE exposure for 1998 for this worker is 985 mrem
- This is a non-emergency situation with no exposure limit extensions authorized

Without exceeding any exposure limits, this worker may remain in this area for a maximum of: (Choices are rounded down to the nearest minute.)

a. 12 minutes

b. 18 minutes

c. 120 minutes

d. 180 minutes

9 Given the following conditions:

- A Peach Bottom Operations Department individual has a current NRC Form 4 on file
- The individual's current yearly Total Effective Dose Equivalent (TEDE) is 1435 mrem
- A General Emergency has been declared
- The Emergency Director has approved a Dose Extension Form to waive the normal PECO administrative dose control limits for all emergency response personnel

Which of the following is the MAXIMUM additional exposure this individual may receive without exceeding any administrative or procedurally based limits? (Assume no other approvals have been received.)

a. 1565 mrem

b. 2565 mrem

c. 3565 mrem

d. 23565 mrem

10 A check-off list (COL) Independent Verification (IV) is required to be completed on 8 system valves located in an area with dose rates of 120 mr/hour.

What is the maximum time available to complete the verification before exceeding the guidelines for Shift Management to consider waiving the IV?

a. 2 minutes

b. 5 minutes

c. 10 minutes

d. 12 minutes

11 A Main Control Room annunciator has a "blue" dot on its window.

Which of the following describes the status of the equipment monitored by that annunciator?

The monitored equipment has a deficiency that:

a. affects the performance of the Transient Response Implementation Plan (TRIP) procedures.
b. is not considered a Main Control Room deficiency.
c. affects the performance of the Emergency Response Procedures (ERP).
d. does not impact any safety related plant equipment.

12 The Control Room has received a report of an injured, potentially contaminated individual in the Unit 2 Reactor Building.

In accordance with the requirements of SE-12, "Injury Response", Medical Response Team call out is made as determined by the:

a. Plant Reactor Operator.
b. Floor Shift Supervisor.
c. Incident Commander.
d. Industrial Risk Management (IRM) representative on call.

13 The Plant Reactor Operator (PRO) has just received a fire alarm from the Turbine Building.

The PRO is REQUIRED to make a call for off-site fire fighting support:

a. after 10 minutes if an actual fire is confirmed.
b. immediately if equipment for safe shutdown is jeopardized.
c. when 2 or more fire alarms are received in the same area.
d. after 20 minutes if the Incident Commander reports the fire is NOT controlled.

14 Given the following conditions:

- Unit 2 is making preparations for a reactor startup from a refueling outage
- Reactor Building ambient temperature is 74 °F
- The Reactor Building Equipment Operator is charging the hydraulic control unit accumulators with nitrogen to a pressure of 590 psig
- Several days later with the Unit at 100% power, Reactor Building temperatures have stabilized at 92 °F

Which of the following describes the expected impact on the Control Rod Drive Hydraulic system operations for these conditions? (Refer to attached figure.)

The individual control rod:

a. normal insertion speeds will be slower and may result in control rod drift alarms.

b. scram speeds will be slower and will result in reduced reactivity addition rates.

c. normal insertion speeds will be faster and may result in "double notching".

d. scram speeds will be faster and may result in mechanism damage.

15 A loss of the "B" Reactor Protection System (RPS) MG Set has just occurred on Unit 3.

Which of the following describes how this will impact the Backup Scram Valves (SV-140A and 140B)?

Loss of the RPS Bus will:

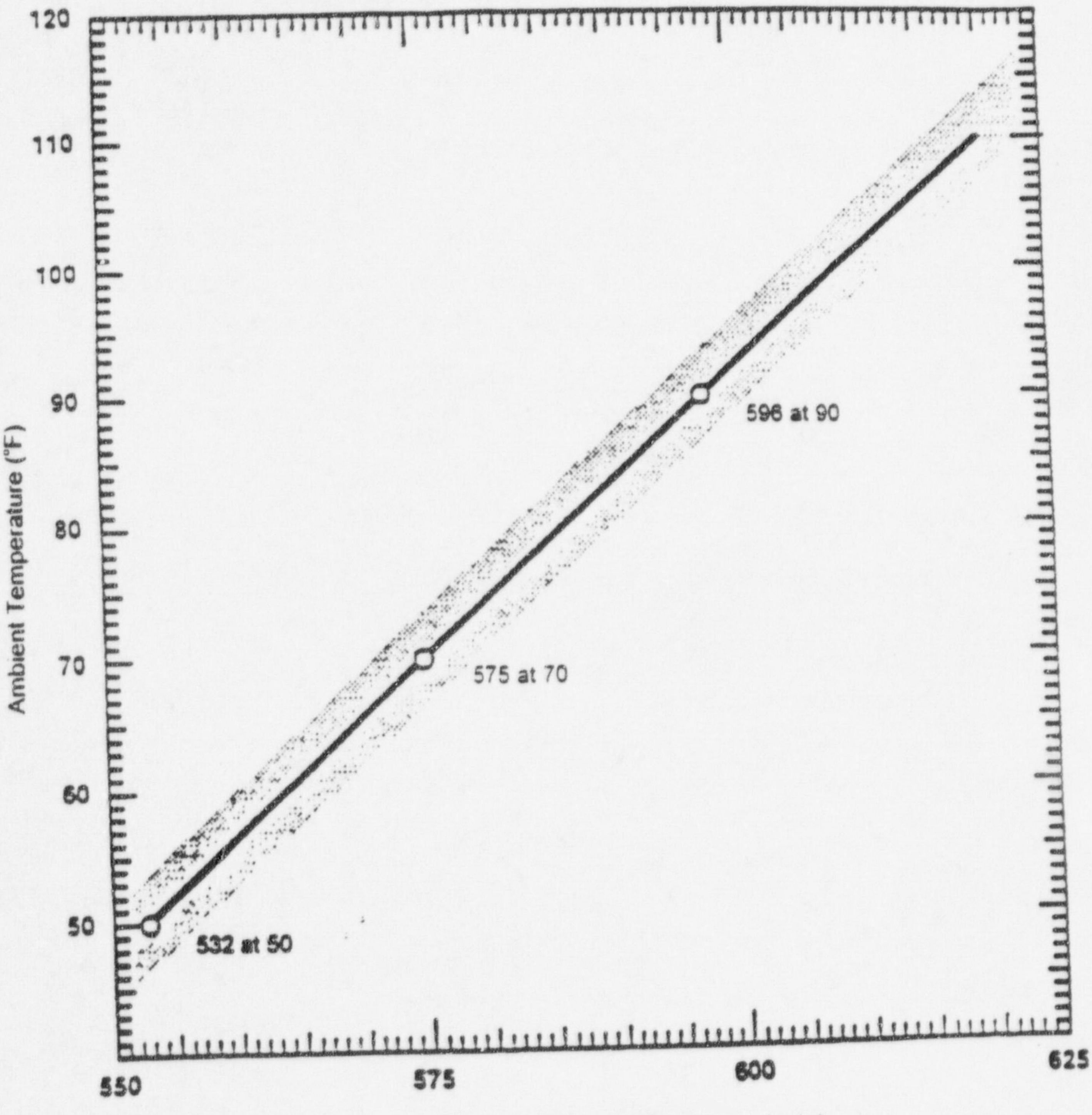
a. not affect the DC power to the Backup Scram Valves and they will remain energized

b. cause a loss of AC power opening two Backup Scram Valves such that a loss of the other RPS Bus will result in a full scram.

c. not affect the DC power to the Backup Scram Valves and they will remain de-energized.

d. cause a loss of AC power opening one of the two Backup Scram Valves required to vent the scram air header.

ACCUMULATOR PRECHARGE NITROGEN PRESSURE VERSES AMBIENT TEMERATURE



16 Control rod 14-31 is resting at position "22" when a single notch withdrawal is initiated by placing the Rod Control Switch in "Notch Out" and then allowing it to spring return to "Off". The Reactor Manual Control System Master Timer fails generating a continuous withdrawal signal.

How does this impact movement of that control rod?

Control rod 14-31 will:

a. be deselected when it reaches its Rod Worth Minimizer withdrawal limit.

b. continue to withdraw until the Operator takes the Rod Select Power Switch to "Off".

c. be deselected immediately by the Master Timer failure circuit.

d. be deselected by the Auxiliary Timer

17 Given the following conditions:

- Unit 2 is operating at 12% power performing a reactor startup
- Control rod 22-51 has a failed reed switch at Notch "08"

Select the expected Rod Worth Minimizer (RWM) response when control rod 22-51 is withdrawn one notch from Notch "06".

The Rod Worth Minimizer will:

a. give only a withdraw error message on control rod 22-51.

b. provide control rod withdrawal blocks on the next selected control rod.

c. provides control rod insert and withdrawal blocks on control rod 22-51only.

d. not allow another control rod to be selected until 22-51 has had a substitute position entered.

18 Given the following conditions on the Unit 3 "A" Reactor Recirculation Pump:

- Seal Staging Flow	0.95 gpm and slowly rising, High flow alarm in
- Seal #1 Pressure	1010 psig and steady
- Seal #2 Pressure	575 psig and slowly rising
- Leak Detection Line	High flow alarm NOT in, flow < 0.1gpm

Which of the following describes what is occurring?

a. The seal staging internal orifice is plugged.

b. Seal #1 is slowly failing.

c. Seal purge flow has been raised.

d. Seal #2 is slowly failing.

19 Given the following conditions:

- Unit 2 was operating at 75% power with a full power rod pattern
- A loss of Main Generator Stator Cooling occurred and was reset in 5 seconds
- The Unit Reactor Operator identified indications of Thermal Hydraulic Instabilities (THI) and scrammed the reactor
- Reactor pressure dropped to 900 psig and has stabilized at 950 psig
- Reactor water level dropped to -38 inches and has been stabilized at +20 inches

What is the status of the Reactor Recirculation System?

a. Both Recirculation Pumps are tripped.

b. The "A" Recirculation Pump is tripped, the "B" Pump is running at 30% speed.

c. Both Recirculation Pumps are running at 30% speed.

d. The "B" Recirculation Pump is tripped, the "A" Pump is running at 30% speed.

20 The Unit 2 Reactor Operator (URO) is starting the "A" Recirculation Pump. After the MG Set Drive Motor Start/Stop switch is placed in "Start" a scoop tube lock occurs. Generator speed when the scoop tube lock occurred was 45% (and lowering to 30% as designed). Assume no additional operator actions.

Which of the following describes the impact on the Recirculation Pump Motor Generator Set for these conditions?

a. The MG Set Drive Motor Breaker will open on exciter field overcurrent.

b. An MG Set lockout will occur on Incomplete Sequence.

c. The MG Set Drive Motor Breaker will open on motor feeder overcurrent.

d. An MG Set lockout will occur on Scoop Tube Lock.

21 Given the following conditions:

- Unit 2 has just completed a power change from 75% to 100% using recirc flow
- Over the next hour, the operators note that power is continuing to slowly rise
- No operator actions are being taken

Which of the following would be the cause of this power rise?

a. Xenon concentration is rising from the power change.

b. Feedwater temperature are rising following the 25% power change.

c. Core inlet subcooling is greater at the higher recirculation flows.

d. Recirculation Pump MG Set oil temperatures are rising.

22 Given the following conditions:

- A power change from 75% to 80% is in progress on Unit 3
- The Unit Reactor Operator (URO) is raising speed on the "B" Recirculation Pump
- While pump speed is rising, the "B" Recirc Pump Speed Control Signal fails downscale

The "B" Recirculation Pump speed change will:

a. stop and local control of the scoop tube will be required to complete the power change.

b. stop, the pump will runback to 30% speed and the scoop tube will lock up.

c. continue until the operator trips or manually locks up the pump.

d. continue until the pump reaches the highest speed set by the URO, then the scoop tube locks up.

23 With Unit 2 and Unit 3 at power, a loss of DC power to the Unit 2 RHR System I logic has occurred. While troubleshooting is in progress, a valid LOCA signal on Unit 2 occurs.

How does this failure and initiation signal impact the Unit 2 and Unit 3 Residual Heat Removal (RHR) System operation?

a. The Unit 2 "B" RHR loop will automatically initiate in the LPCI mode, the Unit 2 "A" RHR loop must be started manually and any running Unit 3 RHR pumps will trip.

b. Both Unit 2 RHR loops will have a normal automatic initiation in the LPCI mode and any running Unit 3 RHR Pumps will automatically trip.

c. The Unit 2 "B" RHR loop will automatically initiate in the LPCI mode, the Unit 2 "A" RHR loop must be started manually and any running Unit 3 RHR pumps must be stopped by the operator.

d. Both Unit 2 RHR loops will have a normal automatic initiation in the LPCI mode and any running Unit 3 RHR Pumps must be stopped by the operator.

24 ST-0-010-306-2, "B" RHR Loop/Pump/Valve/Flow/And Unit Cooler Functional And Inservice Test, is being performed on Unit 2. During the first step of procedure, the "B" Torus Suction Valve (MO-013B) was stroke timed closed. While in this alignment, a valid LOCA signal occurred. The following conditions exist:

- Reactor pressure is 280 psig and lowering slowly
- The "B" RHR Loop Inboard Injection Valve (MO-25B) has opened
- The "D" RHR Pump started and then tripped on overload
- All other systems are operating as designed and no operator actions have been taken

Which of the following describes the "B" LPCI Loop status?

a. The loop is NOT injecting because reactor pressure is above the shutoff head of the running "B" RHR Pump.

b. The loop is injecting after automatically realigning for LPCI injection.

c. The loop is NOT injecting because the "B" RHR Pump is not running.

d. The loop is injecting at 5500 gpm due to current reactor pressure.

25 Given the following conditions:

- Following a transient, Unit 2 is operating in accordance with T-101, "RPV Control"
- The Pressure Control Leg has directed the use of Reactor Water Cleanup (RWCU) in the Recirc Mode
- T-227-2, "Defeating RWCU Isolation Interlocks", has been implemented
- Moments after placing RWCU in the Recirc Mode a "Cleanup Recirc Pump Suction Line Break" alarm is received

Which of the following operator actions would be required to be performed if the system responds as expected?

a. Verify the automatic isolation of the Inboard and Outboard Inlet Valves (MO-15 and 18) and the Outlet Valve (MO-68).

b. Manually close the Inboard and Outboard Inlet Valves (MO-15 and 18) and verify the automatic isolation of the Outlet Valve (MO-68).

c. Manually close the Inboard and Outboard Inlet Valves (MO-15 and 18) and the Outlet Valve (MO-68).

d. Verify the automatic isolation of the Inboard and Outboard Inlet Valves (MO-15 and 18) and manually close the Outlet Valve (MO-68).

26 Given the following conditions:

- Unit 3 has had a complete loss of the E13 4160VAC Bus
- This results in a loss of power to the "A" Residual Heat Removal (RHR) Pump and to the "A" Loop Inboard LPCI Injection Valve (MO-25A)
- A valid LOCA signal occurs

What must occur to result in a final, design injection flowrate for these conditions of 30,000 gpm.

a. The RHR Loop Cross-Tie Valve (MO-20) must be unlocked and opened by an operator.

b. An operator must manually transfer the Inboard LPCI Injectior Valve (MO-25A) to the alternate power supply.

c. The Outboard LPCI Injection Valve (MO-154A) must automatically open to inject though the normally open MO-25A.

d. The Inboard LPCI Injection Valve (MO-25A) must automatically transfer to the alternate power supply.

27 Given the following conditions:

- Unit 2 is shutdown with fuel handling operations in progress
- The "A" Loop of Residual Heat Removal (RHR) is operating in the Shutdown Cooling Mode
- A leak occurs between the Shutdown Cooling Suction Valves (MO-17 and 18) rapidly lowering reactor water level and resulting in a LPCI initiation signal on Lo-Lo-Lo level
- All expected actions occur

What is the expected response of the Shutdown Cooling System and reactor water level for these conditions? (Do not consider Core Spray in your selection of answers.)

a. Shutdown cooling will isolate. Reactor level will stabilize but not recover unless operator action is taken to inject.

b. Shutdown cooling will NOT automatically isolate. Operator actio` is required to isolate the leak and inject with RHR to recover level.

c. Shutdown cooling will isolate. Reactor level will rise due to the "B" Loop of RHR injecting in the LPCI mode until stopped by the operator.

d. Shutdown cooling will NOT automatically isolate. Operator action is required to isolate the leak allowing automatic LPCI injection to recover level.

28 Given the following conditicns:

- The Unit 2 High Pressure Coolant Injection (HPCI) system was manually initiated using the SO procedure due to lowering reactor water level
- The HPCI Manual Initiation Pushbutton was used
- The HPCI Manual Initiation Pushbutton stuck in the "Armed" position
- Water level has been returned to the normal band and is stable

Which of the following is the correct method to shutdown HPCI in accordance with SO 23.2.A-2, "HPCI System Shutdown"?

a. Depress and hold the HPCI System Remote Trip Pushbutton until turbine speed indicates "0" rpm, then place the Aux Oil Pump in "Pull-To-Lock".

b. Depress the Local Manual Trip on the HPCI Turbine pedestal.

c. Depress and hold the HPCI System Remote Trip Pushbutton, close the Steam Admission Valve (MO-14) and then release the pushbutton.

d. Depress the "HPCI Isolation" pushbutton.

29 With the High Pressure Coolant Injection (HPCI) operating and injecting to the reactor vessel a loss 480 VAC power supplying HPCI components occurs.

What is the expected result of this failure?

a. Loss of speed control resulting in a trip of HPCI on overspeed

b. Inability to remove gland seal non-condensables causing high airborne conditions in the HPCI Room.

c. Inability to isolate a leak on the HPCI steam line immediately outside the drywell.

d. Loss of the Auxiliary Oil Pump on HPCI shutdown resulting in damage to the HPCI pump and turbine bearings.

30 Given the following conditions:

- The Unit 2 "System II Core Spray to Top Of Core Plate High Diff Pressure" alarm has been received
- Reactor pressure is 300 psig
- Drywell pressure is 0.35 psig and steady

With the "B" and "D" Core Spray Pumps running, system flow will be going into:

a. the area inside the core shroud on top of the core but bypassing the Core Spray sparger.

b. the downcomer area between the core shroud and the reactor vessel wall.

c. the drywell area between the maintenance valve and the reactor vessel wall.

d. the drywell area between the Core Spray maintenance valve and the testable check valve.

31 The Unit 3 "A" Core Spray Pump was running in the test return mode for a surveillance when a plant transient resulted in the following conditions:

- A leak into the drywell results in pressure rising to 2.15 psig
- Reactor water level is -162 inches
- Reactor pressure is 690 psig and lowering

Select the expected response for these conditions?

a. The "A" Core Spray Pump trips, the Test Return Valve (MO-26A) closes, the pump restarts and the Injection Valves (MO-11 & 12) open.

b. The "A" Core Spray Pump continues to run on minimum flow, the Test Return Valve (MO-26A) closes and the Injection Valves (MO-11 & 12) open.

c. The "A" Core Spray Pump continues to run on minimum flow, the Test Return Valve (MO-26A) closes and the Injection Valves (MO-11 & 12) do NOT reposition.

d. The "A" Core Spray Pump trips, the Test Return Valve (MO-26A) closes, the pump restarts and runs on minimum flow and the Injection Valves (MO-11& 12) do NOT reposition.

32 Given the following conditions:

- A failure-to-scram (ATWS) has occurred on Unit 2
- The "A" Standby Liquid Control (SBLC) Pump was started 5 minutes ago
- Initial SBLC Storage Tank level was 74%
- Reactor pressure is 1050 psig

Which of the following is NOT an expected SBLC system indication for these conditions?

a. "A" SBLC Pump discharge pressure is 1450 psig.

b. Group II/III Inboard and Outboard isolation relays not reset alarms are in

c. The Squib Valve Continuity lights (amber) are illuminated.

d. Core thermal power is slowly lowering.

33 Given the following conditions:

- Unit 3 experienced a Group I isolation
- The reactor scrammed and all normal scram actions were completed
- The Scram Discharge Volume Keylock Switch has been placed in "Bypass"

Which of the following would prevent resetting the scram?

a. Condenser vacuum is 18 inches.

b. The Main Steam Isolation Valves are all closed.

c. Reactor level is "0" inches.

d. Reactor pressure is 1075 psig.

34 Unit 2 is operating at full power. While transferring the "A" Reactor Protection System (RPS) Bus from the Normal to the Alternate power supply on Unit 2, the Transfer Switch malfunctions and "Alternate" power is unable to be selected. It is decided to leave the bus deenergized for initial troubleshooting of the switch.

Which of the following is an RPS consideration with this bus deenergized for a long time period?

a. Continuous operation with one of the Backup Scram Valves open.

b. A possible loss of redundancy with the "one-out-two-taken-twice" logic bypassed.

c. Continuous operation with two of the four ARI valves open.

d. A possible control rod block and eventual reactor scram on Scram Discharge Volume high levels.

35 Which of the following are the indications for a coupled control rod that has drifted full out?

a. Double dashes ("--") and a Rod Overtravel annunciator

b. Digital "48" with a red backlight

c. Digital "48" and a Rod Overtravel annunciator

d. Double dashes ("--") with red backlight.

36 With Unit 2 at 100% a transient requiring control rod insertions per GP-9-2, "Fast Reactor Power Reduction", occurs. While in progress the "RPIS Inoperative" alarm is received. The Unit Reactor Operator stops driving rods to investigate the annunciator.

Continued control rod insertions are possible:

a. only by placing the Mode Selector Switch in "Shutdown".

b. for all control rods with the exception of the rod with the RPIS failure.

c. by using the Emergency In/Notch Override Switch in "Emergency In".

d. by using the Rod Control Switch held in the "In" position.

37 A Traversing Incore Probe (TIP) trace is in progress on Unit 2 when a high drywell pressure occurs. After checking the TIP Valve Control Monitor the Plant Reactor Operator reports the following indications (Refer to attached figure):

- "Squib Monitor" lights - both extinguished
- "Shear Valve Monitor" lights - both extinguished
- "Ball Valve Open" lights - both illuminated
- "Ball Valve Closed" lights - both extinguished

Which of the following describes the status of the TIP system and the required operator actions?

a. The system has responded as designed. Operator action is required to close the ball valves.

b. The TIP detectors may not have withdrawn. Withdraw the detectors and verify the ball valves close.

c. The system has responded as designed. Operator action is required to fire the shear valves.

d. The TIP detectors may not have withdrawn. Fire the shear valves, withdraw the remaining cable and close the ball valves.

38 With Unit 2 operating at 75% power, the Rod Block Monitor (RBM) gain change circuit malfunctions and does NOT provide any LPRM input signal gain adjustments.

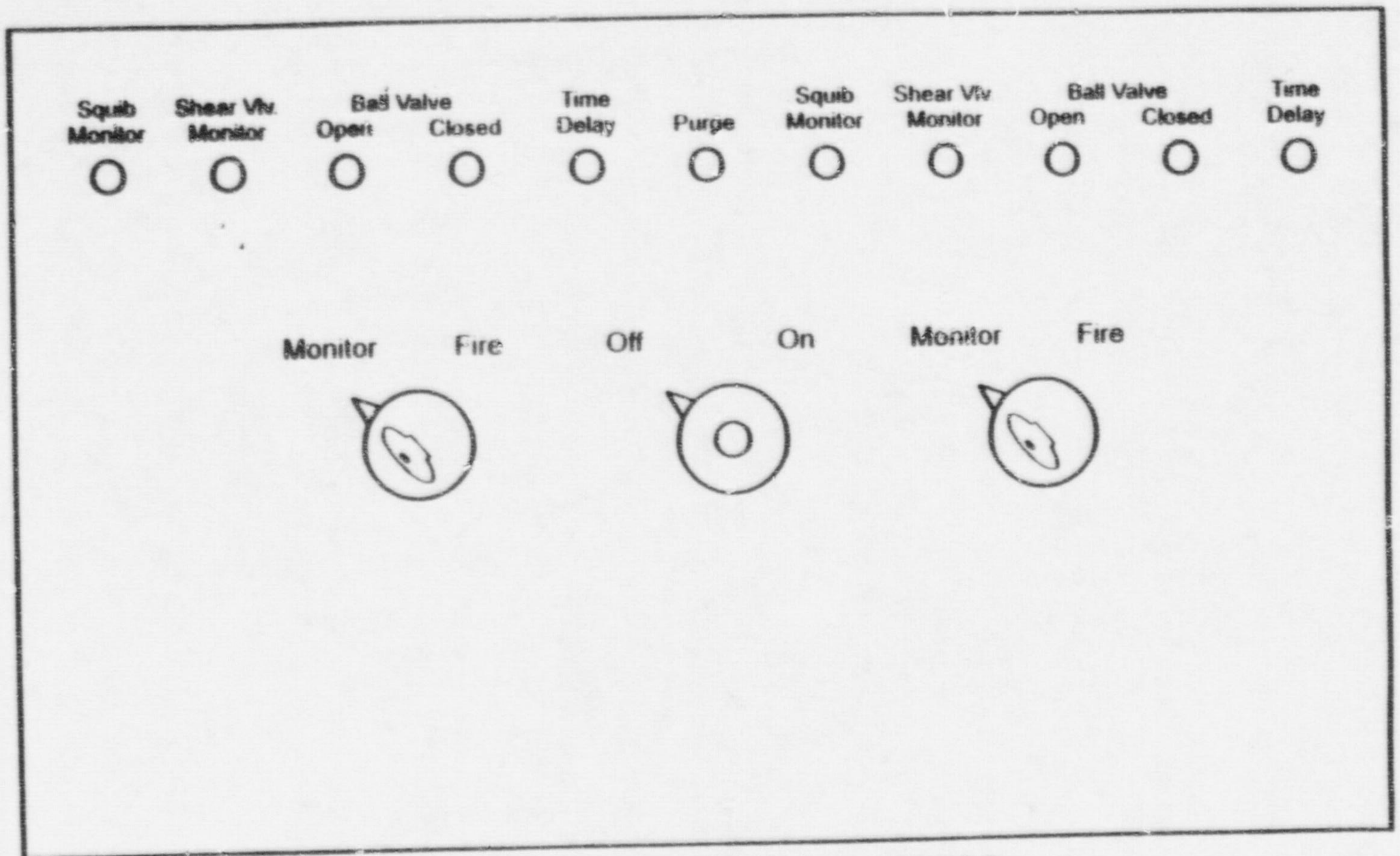
Which of the following describes how this may impact control rod withdrawals?

a. The RBM would shift to the low trip setpoint and generate a rod block.

b. The RBM would transfer from the normal to the alternate APRM channel on the failure to null.

c. The RBM would initiate a select block when the next control rod is selected if reference power above 30%.

d. Localized core power could be higher on individual control rod withdrawals before any protective actions occur.



TIP VALVE CONTROL MONITOR

39 At what point during a Unit 3 reactor startup can the Unit Reactor Operator expect a significant increase in Wide Range Neutron Monitoring (WRNM) nuclear instrumentation response during control rod withdrawals?

a. Control rod withdrawals made after steam is being drawn from the reactor.

b. Control rod withdrawals made as reactor power passes 1.00E0% on WRNMs.

c. Withdrawal of a center control rod during a fast recovery startup.

d. Initial rod withdrawals from 50% rod density in the startup.

40 Given the following conditions on Unit 3:

- A reactor startup is in progress
- The Mode Selector Switch is in "Startup/Hot Standby"
- All WRNM Channels are reading 2 counts per second
- All APRM Channels are reading "downscale"
- The Rod Block Monitor is reading "downscale"
- Rod Select Power is "on"
- All systems are operating as designed

Which of the following describes why control rod withdrawals are not permitted?

a. An Average Power Range Monitoring Rod Block

b. A Rod Block Monitor Rod Block

c. A Wide Range Neutron Monitoring Rod Block

d. A Rod Select Power On - No Rod Selected Rod Block

41 Given the following conditions:

- A failure-to-scram (ATWS) has occurred on Unit 2
- Reactor power 22% and steady
- Reactor water level +10 inches and steady
- Reactor pressure 1010 psig and steady
- Drywell pressure 1.7 psig and rising slowly
- Scram Discharge Volume Vent and Drain Valves are closed
- Scram Discharge Volume High Level Scram is in
- Full core display blue scram lights are all illuminated
- Mode Selector Switch is in "Shutdown"

What actions must be taken before the scram can be reset to allow draining the Scram Discharge Volume for these conditions?

a. The Reactor Protection Trips must be bypassed using jumpers.

b. The SDV High Level Scram Bypass Switch must be placed in "Bypass"

c. The Mode Selector Switch must be taken out of "Shutdown" and the SDV High Level Scram Bypass Switch placed in "Bypass"

d. An immediate scram reset is possible after the Mode Selector Switch in "Shutdown" scram signal time delay has expired.

42 Given the following conditions:

- A Unit 2 startup is in progress with power at 20%
- Recirculation flow is 30%
- The "A" APRM Flow Unit output remains at 30% as recirculation flow is raised

As the plant startup continues, what will be the FIRST protective action to occur and the reason for that action?

a. A control rod block will occur due to flow biased neutron flux upscale.

b. A control rod block will occur due to a flow unit comparator trip.

c. A half scram will occur due to a flow unit "inop" signal.

d. A [illegible] scram will occur due to flow biased neutron flux upscale.

43 During steady power reduction from 100% to 65% power on Unit 3 the Unit Reactor Operator notes Wide Range reactor water level indications, which had been reading about 10 inches less than Narrow Range, are slowly rising. Actual reactor water level remains unchanged.

Which of the following describes what is occurring?

a. The density compensation signal (reactor pressure) has failed full "downscale".

b. The density compensation signal (reactor pressure) is lowering as power is reduced resulting in a lowering d/p on the level instrument, therefore an indicated level rise.

c. The Digital Feedwater redundant feedback signals have failed full "upscale".

d. The reduction in recirculation flow is raising the pressure at the variable leg tap resulting in a lowering d/p on the level instrument, therefore an indicated level rise.

44 Given the following conditions:

- Unit 2 has experienced a loss of all AC power (station blackout)
- The Reactor Core Isolation Cooling (RCIC) system automatically initiated
- Reactor water level is now -52 inches and rising
- The Control Room Supervisor directs the Unit Reactor Operator to isolate RCIC

What will be the expected RCIC system response when the operator depresses the Manual Isolation Pushbutton?

a. A normal RCIC system isolation and turbine trip will occur.

b. A RCIC turbine trip and system isolation will occur except the Inboard Steam Isolation Valve (MO-15) will not close.

c. No RCIC isolation actions or turbine trip will occur.

d. A RCIC turbine trip and system isolation will occur except the Outboard Steam Isolation Valve (MO-16) will not close.

45 Unit 2 requires an Emergency Blowdown after performing steam cooling in T-111, "Level Restoration". All actions required by T-112, "Emergency Blowdown", have been taken but only 3 Safety Relief Valves (SRV) can be opened and no other means of depressurization is available.

Which of the following describes the consequences of this failure?

a. Steam removal rate from the core is not adequate to ensure adequate decay heat removal exists.

b. The pressure reduction rate will not allow low pressure injection systems to inject soon enough to recover level before core uncovery occurs.

c. Steam removal rate during a LOCA is not adequate to prevent exceeding the drywell design pressure.

d. The pressure reduction rate will not allow low pressure injection systems to inject prior to reaching the Minimum Steam Cooling RPV Water Level.

46 Given the following conditions:

- Unit 3 is in Mode 4 for a scheduled 6 day maintenance outage on drywell cooling
- A loss of shutdown cooling has occurred
- Shutdown cooling will not be restored for several hours
- Reactor water level +23 inches and steady
- Reactor water temperature 195 degrees F and rising

Select the ON-125, "Loss of Shutdown Cooling", action REQUIRED for these conditions.

a. Establish a vent path to the drywell via the reactor head vents.

b. Raise reactor water level to 30 inches to promote natural circulation.

c. Enter and take actions in accordance with T-102, "Primary Containment Control".

d. Verify the primary containment is in an Operable condition in accordance with GP-2.

47 Given the following conditions:

- A reactor scram has occurred on Unit 2
- T-101, "RPV Control", has been entered
- Subsequent to the scram, panel 20Y33 is lost

How will this failure impact the pressure control options in T-101?

a. Safety Relief Valve solenoid power will be lost making the SRVs unavailable for pressure control.

b. The Inboard Main Steam Isolation Valves will close making the main condenser unavailable for pressure control.

c. The Reactor Water Cleanup System will isolate and will be unavailable for pressure control.

d. The High Pressure Coolant Injection Inboard Steam Isolation Valve (MO-15) will close making HPCI unavailable for pressure control.

48 Following a large leak, the following conditions exist on Unit 2:

- Drywell pressure 12 psig
- Drywell temperature 225 degrees F
- Reactor water level -100 inches
- Reactor pressure 290 psig
- All Residual Heat Removal Pumps are running
- All drywell pressure signal inputs to RHR logic have failed to "0" psig
- The CRS has verified that plant conditions meet the Drywell Spray Initiation Curve

Under these conditions, containment sprays are:

a. not available.

b. available after the Containment Spray Valve Control Switch is placed in "Manual".

c. available after the Containment Spray Valve Reset pushbutton is pressed.

d. available after the Override 2/3 Core Coverage key lock switch is placed in "Override".

49 Given the following conditions:

- Unit 2 is in Mode 5
- The Mode Selector Switch is in "Refuel"
- The Refueling Platform is over the spent fuel pool
- A fuel bundle has been loaded on the Main Hoist and raised out of the fuel pool storage rack
- The Unit Reactor Operator has just received a control rod block

What additional action was taken to cause this rod block?

a. The Refueling Platform operator raised the Main Hoist to the "full up" position.

b. The Unit Reactor Operator placed the Mode Selector Switch in "Startup/Hot Standby".

c. The Refueling Platform operator moved the platform over the reactor vessel.

d. The Unit Reactor Operator has selected, but NOT withdrawn, a single control rod.

50 Given the following conditions:

- Unit 3 is operating at 50% power
- Instrument nitrogen has been lost
- The instrument air backup valves to instrument nitrogen did NOT open
- DC power has been lost to the MSIV pilot solenoids
- No operator actions have been taken

What will be the expected Main Steam Isolation Valve (MSIV) response? (NOTE: Consider ONLY the response of the MSIVs for this question.)

a. The Inboard MSIVs will eventually drift closed, the Outboard MSIVs will remain open.

b. All eight MSIVs will remain open.

c. The Outboard MSIVs will eventually drift closed, the Inboard MSIVs will remain open.

d. All 8 MSIVs will close.

51 Unit 2 is operating at 100% power with all systems operating as designed. A failure in the Maximum Combined Flow Limit potentiometer causes its setpoint to lower to 80%.

Which of the following is the expected INITIAL response of the Electro-Hydraulic Control System for this failure?

a. The Main Turbine will trip from a power/load imbalance.

b. The Turbine Control Valves will close to a steam flow of 80%. The Turbine Bypass Valves will open to a steam flow of 20%.

c. The Load Set potentiometer will runback to maintain turbine load matched with steam flow.

d. The Turbine Control Valves will close to a steam flow of 80%. The Turbine Bypass Valves will remain closed.

52 Given the following conditions:

- Unit 2 is operating at 100% power
- Main generator load is 1150 MWe
- Power factor is 0.95 lagging
- A generator leak has resulted in a stable hydrogen pressure of approximately 60 psig

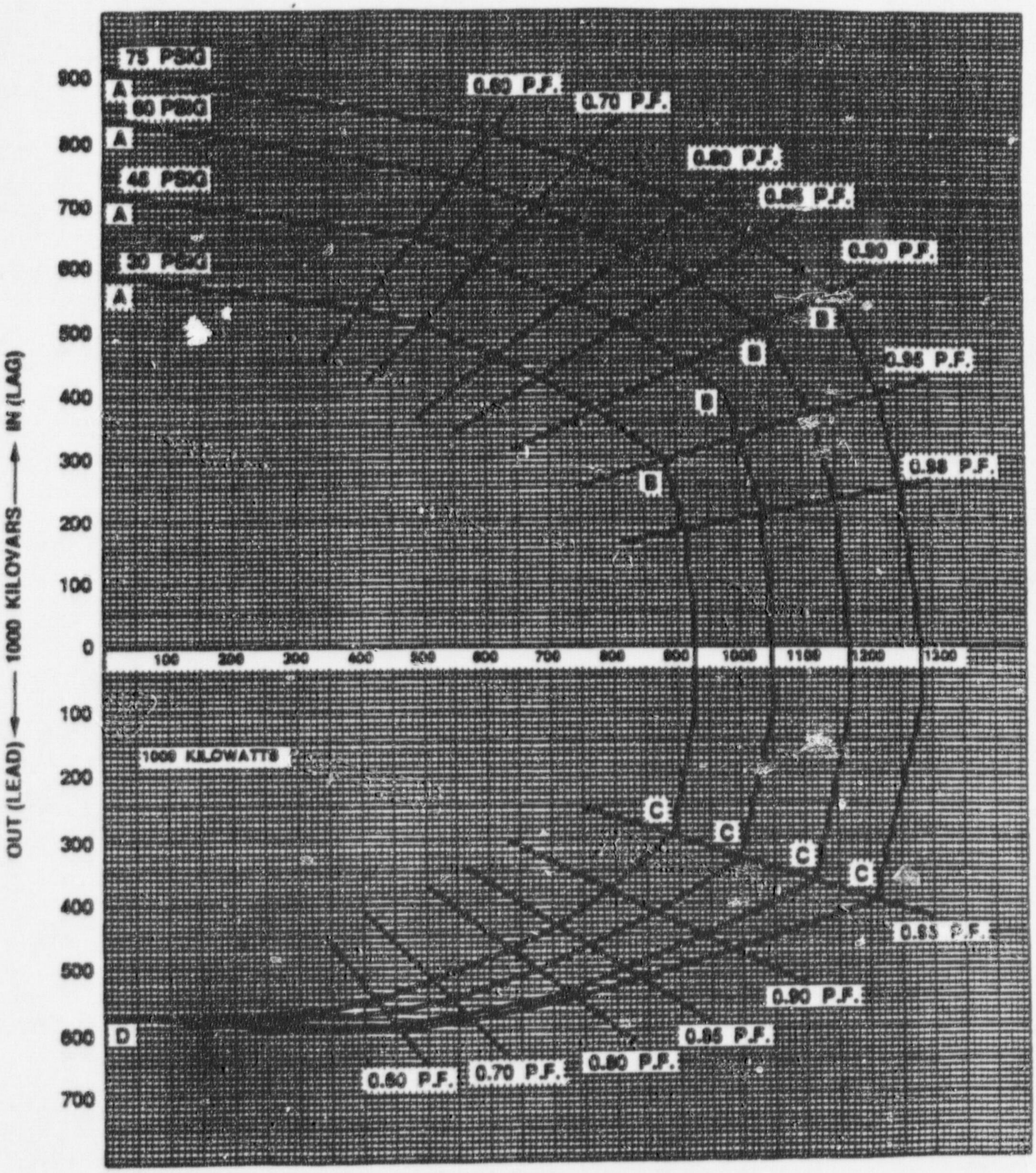
Select the maximum main generator load allowed for these conditions. (Refer to attached figure.)

a. 1070 megawatts

b. 1110 megawatts

c. 1150 megawatts

d. 1190 megawatts



75 PSIG
60 PSIG
45 PSIG
30 PSIG
A
B
C
D
0.60 P.F.
0.70 P.F.
0.80 P.F.
0.85 P.F.
0.90 P.F.
0.95 P.F.
0.98 P.F.
1000 KILOWATTS
OUT (LEAD) ← 1000 KILOVARS → IN (LAG)

53 Given the following conditions:

- Unit 2 is operating at 100%
- All plant systems have been operating as designed
- Reactor Feedwater Pump (RFP) speeds are all approximately 4150 rpm
- A transient occurs causing a single Condensate Pump trip
- Concurrent with this trip, a loss of the "B" RFP control signal occurs

Which of the following describes the final status of the RFPs for these conditions?

a. "A" and "C" RFP speed will remain constant and the "B" RFP maximum speed will be set by a limiter.

b. "A", "B" and "C" RFP maximum speeds will be set by a limiter.

c. "A" and "C" RFP maximum speed will be set by a limiter and the "B" RFP speed will remain constant.

d. "A", "B" and "C" RFP will all receive a continuous lower speed signal.

54 Which of the following describes how the Reactor Building Ventilation/Standby Gas Treatment (SBGT) Systems operate with SBGT in service under non-emergency conditions (no Group III isolation signal present)?

With no Group III isolation signal present:

a. the SBGT Fan Vortex Dampers are throttled as needed from the control room.

b. the Reactor Building and Refuel Floor to SBGT Flow Dampers (PO-20477-1 & 2) are full open.

c. the SBGT Fan Vortex damper positions to automatically control pressure.

d. the Reactor Building and Refuel Floor to SBGT Flow Dampers (PO-20477-1 & 2) are throttled locally to control pressure.

55 Given the following conditions:

- Unit 2 is operating at 75% power
- The "A" Core Spray Pump is running for a surveillance
- The E212 Breaker has just opened
- All plant systems respond as designed

The "A" Core Spray Pump:

a. continue to run if bus voltage remains greater than 30% during the fast transfer.

b. will trip on undervoltage but will automatically restart if the Core Spray system receives an initiation signal.

c. continue to run if the fast transfer is completed in less than 0.25 seconds.

d. will trip on undervoltage and the breaker control switch must be placed to "Trip" and then to "Close" to restart the pump.

56 Given the following conditions:

- Both Units are operating at 100% power
- All systems are operating as designed in their normal lineups
- While applying a clearance the Plant Reactor Operator (PRO) opens the E-212 breaker

Which of the following describes the actions that must occur to reenergize the E-12 4KV Bus?

a. The PRO will have to close the E-312 breaker.

b. The E-1 Diesel Generator will automatically start and the E-12 Output Breaker will close.

c. The E-312 breaker will close provided the E-212 breaker switch remains in the "Normal After Close" position.

d. The E-1 Diesel Generator will automatically start and the PRO will have to close the E-12 Output Breaker.

57 Given the following conditions:

- The Unit 3 - 3D 125/250 Volt Battery Charger is in service and providing a normal "float" charge on its battery
- While in this lineup, AC power to the charger is lost
- The bus supplying the charger is reenergized after 20 minutes by its associated diesel generator

Which of the following describes the expected response of this battery charger?

The 3D Battery Charger will:

a. return to the "float" mode to trickle charge the battery.

b. trip and is interlocked "off" with the diesel generator powering the bus.

c. reset to the "equalize" mode to recharge the battery.

d. trip and must be manually restored as permitted by diesel generator loading.

58 Given the following conditions:

- The E-42 4KV Bus has lost power
- The fast transfer and Diesel Generator start both failed to occur automatically
- The E-4 Diesel Generator (DG) was started with the "Quick Start" pushbutton
- The E-42 breaker is closed and the DG is now carrying all the loads on the E-42 4KV Bus

Which of the following describes the current Mode of operation of the DG and what is required synchronize the DG back to the Grid?

The E-4 DG is operating in:

a. Parallel, the DG Quick Start pushbutton must be pressed again and synch must be completed within 3 minutes.

b. Unit, the DG Quick Start pushbutton must be pressed again and synch must be completed within 3 minutes.

c. Parallel, the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes.

d. Unit, the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes.

59 Given the following conditions:

- Unit 2 is operating at 10% power
- The E-3 Diesel Generator (DG) Start Mode Selector Switch is in "Auto"
- The E-3 DG Control Switch is in "Pull-To-Lock" for a test
- The E-32 4KV Bus has lost power with a failure to Fast Transfer
- A loss of all drywell cooling has resulted in a 2.5 psig drywell pressure

What is the status of the E-3 DG for these conditions?

a. Running with the E-32 Output Breaker open

b. Tripped with normal post-trip alarms in

c. Running with the E-32 Output Breaker closed

d. Ready for a start with the "Not In Auto" alarm in

60 Which of the following describes how the Off-Gas system charcoal beds reduce the radioactive effluents released by the plant?

a. Nitrogen gas isotopes from the disassociation of water are removed using holdup and decay.

b. Fission product noble gas isotopes are removed using holdup and decay.

c. Nitrogen gas isotopes from the disassociation of water are removed via ion exchange.

d. Radioactive particles are removed via ion exchange.

61 Given the following conditions:

- Unit 2 is performing a reactor startup
- Reactor power is 3%
- The Mode Selector Switch is in "Startup/Hot Standby"
- The Main Steam Isolation Valves (MSIV) are open
- The "A" Main Steam Line Radiation Monitor has failed "downscale"
- Tech Specs have not yet been evaluated for this failure

Select the conditions that will result in a complete MSIV closure.

a. The "C" and "D" Main Steam Line Radiation Monitors fail "upscale".

b. The "B" Main Steam Line Radiation Monitor fails "upscale".

c. The "B" and "D" Main Steam Line Radiation Monitors fail "upscale" and the Mode Selector Switch is placed in "Run".

d. The "C" Main Steam Line Radiation Monitor fails "downscale" and the Mode Selector Switch is placed in "Run".

62 During a lowering instrument air pressure transient ON-119, "Loss Of Instrument Air", directs a rapid plant shutdown if pressure is not stable above 75 psig.

How will a continued loss of instrument air impact the plant if it stays at power?

a. The chilled water to the Drywell Fans will be lost causing a drywell pressure rise.

b. The Reactor Building Ventilation system will no longer support normal building heat loads.

c. The Inboard Main Steam Isolation Valves will fast close (3 to 5 seconds).

d. The operability of plant emergency core cooling injection systems will be affected.

63 Given the following Unit 2 parameters during a normal shutdown with both Recirculation Pumps operating:

	1400	1430
- Reactor pressure	485 psig	325 psig
- Recirc Loop A/B temp	435 deg F	417 deg F
- Rx vessel shell temp	465 deg F	442 deg F
- Rx vessel flange temp	463 deg F	450 deg F
- Bottom head drain temp	402 deg F	378 deg F

What is the cooldown rate for these conditions?

a. 36 degrees F/hour

b. 46 degrees F/hour

c. 78 degrees F/hour

d. 88 degrees F/hour

64 Which of the following describes how to place the Main Control Room Emergency Ventilation System (MCREV) in service from the Control Room?

a. Both MCREV System initiate pushbuttons must be pressed simultaneously.

b. Two of the four Control Room fresh air supply duct radiation monitors must be placed in "Test".

c. The fans and dampers must be manipulated via the Control Room controls.

d. The running Control Room Fresh Air Supply fan must be stopped.

65 Given the following conditions:

- Unit 2 was operating at 75% power when the "A" Recirculation Pump tripped
- Reactor power is 48%
- Calculated Total Core flow is 36%
- The Immediate Operator Actions of OT-112, "Recirculation Pump Trip" have been completed

Select the required operator actions for these conditions. (Power/Flow Map is attached)

a. Raise "B" Recirculation Pump speed to reach 45% core flow.

b. Lower the "B" Recirculation Pump speed to lower thermal power to less than 34%.

c. Restart the "A" Recirculation Pump to reach 50% core flow.

d. Scram and enter T-100, "Scram".

66 Unit 2 plant parameters have stabilized following a trip of the "A" Recirculation Pump from 100% power. The "B" Recirculation Pump is running at 1150 rpm and indicated core flow is 40%.

Which of the following describes the validity of total indicated core flow for these conditions?

Total indicated core flow:

a. is accurate if the "B" Recirculation Pump speed is greater than 40%.

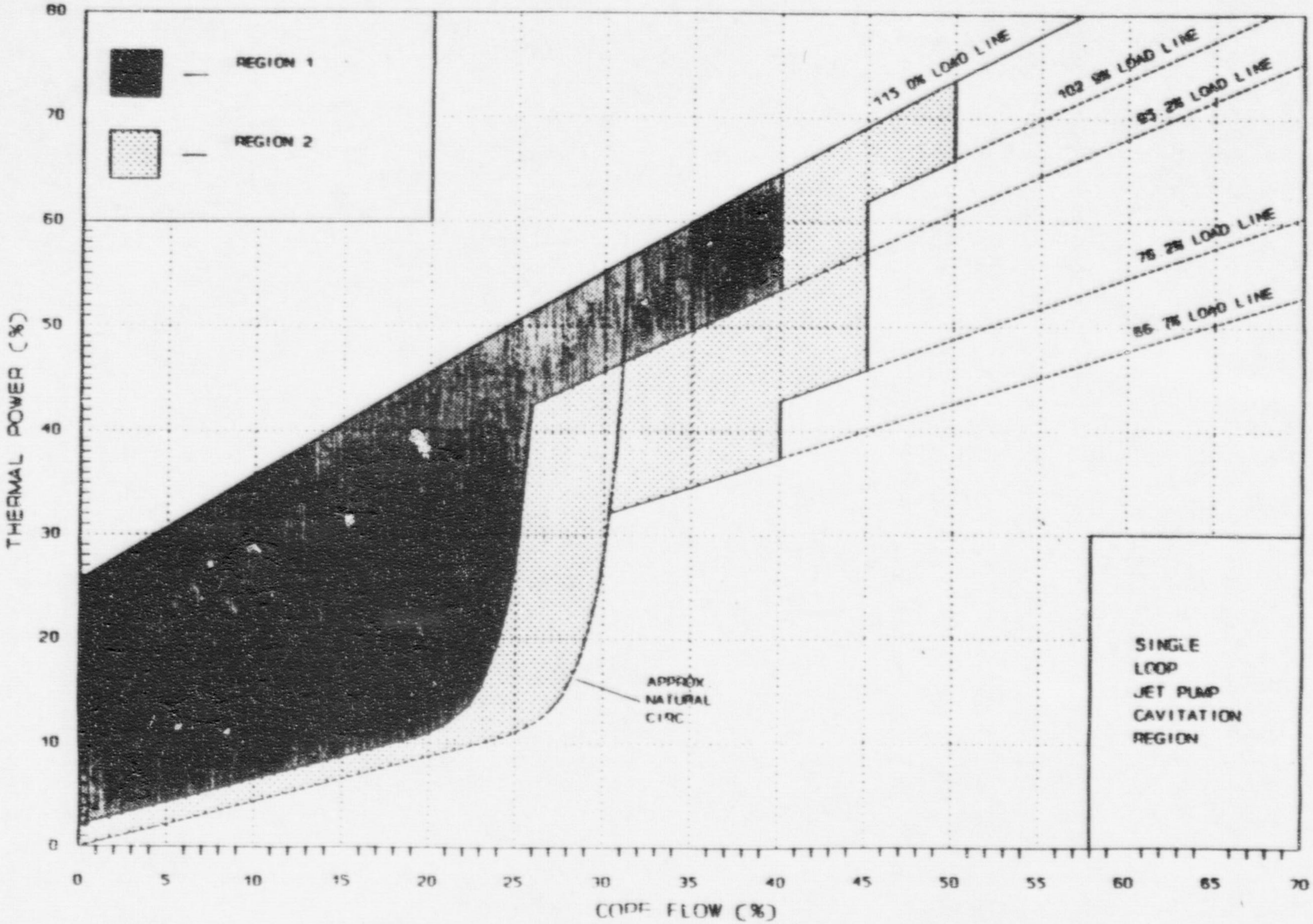
b. must be raised by two times the "A" Recirculation Loop indicated flow.

c. is accurate if the "A" Recirculation Loop indicated flow is less than 3.0 Mlbs/hr.

d. must be reduced by two times the "A" Recirculation Loop indicated flow.

PBAPS POWER FLOW OPERATION MAP

FIGURE 1



67 Given the following conditions:

- Unit 2 is at 2% power with the Mode Selector Switch in "Startup/Hot Standby"
- Main turbine shell warming is in progress
- The "C" Reactor Feedwater Pump is feeding the reactor
- While transferring from the Mechanical Vacuum Pump to the Steam Jet Air Ejectors, main condenser vacuum begins to lower
- No operator actions are taken

What will be the cause of the reactor scram?

a. Turbine Control Valve fast closure

b. Lowering Main Condenser vacuum (direct RPS trip)

c. Low reactor water level

d. High reactor pressure

68 Given the following conditions:

- Unit 2 was operating at 100% power
- Annunciator 220 F-5, "Inverter Trouble", was received indicating a loss of 20Y050
- The reactor was later scrammed and the turbine tripped

Which of the following is the reason why this failure requires reactor pressure control via the Safety Relief Valves?

The static inverter loss will:

a. cause a full Group I Main Steam Isolation Valve closure.

b. cause a "full open" signal to the Turbine Bypass Valves requiring the EHC Pumps to be tripped to prevent a rapid depressurization.

c. result in a closure of the Inboard Main Steam Isolation Valves.

d. result in a loss of Turbine Bypass Valve opening capability.

69 Given the following conditions:

- Unit 3 is at 2% power, performing a startup
- Main turbine shell warming is in progress
- The "C" Reactor Feedwater Pump is feeding the reactor
- A loss of both divisions of 24 VDC power occurs causing a reactor scram

How does this power failure result in a scram?

a. The MSIVs went closed.

b. Wide Range Neutron Monitoring nuclear instrumentation is inoperable.

c. The Main Turbine tripped

d. The "C" Reactor Feedwater Pump tripped.

70 Given the following conditions on Unit 2:

- With power at 22%, a loss of Stator Cooling occurred
- All automatic actions occurred as designed
- The turbine did not trip
- The Immediate Operator Actions of OT-113, "Loss Of Stator Cooling", have been completed
- There is no time estimate for restoration of Stator Cooling

The decision on if and when to trip the Main Turbine is based upon:

a. stator cooling water conductivity at the start of the transient.

b. the rate of increase of stator temperatures after the runback is complete.

c. the current plant location on the power to flow map.

d. final main generator field (amps) after the runback has gone to completion.

71 Given the following conditions on Unit 2:

- A scram has occurred due to a loss of feed
- Reactor water level reached -35 inches and is recovering using HPCI
- Nine control rods did not insert and remain at Notch "48"
- The Control Room Supervisor has entered T-100, "Scram" and T-99, "Post Scram Restoration"

Which of the following describes when the Unit Reactor Operator shall initiate ARI?

ARI shall be initiated:

a. as part of the expected actions directed by OM-P-16.1:5, "OSPS Reactor Operator Response To Reactor Scram".

b. as part of the "verify the scram" actions of T-100, "Scram".

c. when directed by T-99, "Post Scram Restoration".

d. when directed by GP-4, "Manual Reactor Scram".

72 Given the following conditions on Unit 2:

- A Main Steam Isolation Valve closure from 100% power has occurred
- Reactor pressure is 1050 psig
- Instrument nitrogen is not available to the Safety Relief Valves (SRVs)
- The pressure control leg of T-101, "RPV Control" requires a cooldown rate of less than 100 degrees F/hour and directs depressurization by "prolonged" SRV openings

Which of the following describes how the SRVs should be operated for these conditions?

The SRVs should be manually opened:

a. until 350 psig reactor pressure and then returned to the "Close" or "Automatic" position.

b. and closed to obtain pressure reductions in increments of slightly less than 100 psig until the reactor is depressurized.

c. and maintained open to obtain pressure reductions slightly less than the equivalent of 100 degrees F before shutting.

d. and maintained open until the reactor is depressurized or until pneumatic pressure is no longer available.

73 During a high reactor water level condition, the operator is directed to utilize LI-2(3)-2-3-86 to determine if main steam line flooding is occurring.

Which of the following must done to ensure this level indicator is providing an accurate reactor water level?

a. The indicated LI-86 level must be confirmed by an independent level indicator or computer point.

b. Reactor pressure must be compared to indicated level.

c. The drywell temperature and reactor pressure must be confirmed to be above the RPV Saturation Curve.

d. An adjustment must be made to account for recirculation pump flows.

74 Given the following conditions:

- Unit 2 is operating at 85% power
- All three Reactor Feedwater Pumps (RFP) are operating in "Automatic"
- A Feedwater Master Level Controller malfunction is causing a 5 inch per minute reduction in reactor water level
- Reactor water level is +16 inches

What are the Immediate Operator Actions required by OT-101, "Reactor Low Level"?

a. Verify Recirculation Pump runback to 45%.

b. Enter and take the actions as directed by GP-4, "Manual Reactor Scram".

c. Place the Mode Selector Switch in "Shutdown".

d. Take manual control of feedwater and attempt to restore level.

75 Unit 2 is operating at 100% when the DCC-X Digital Feed Control System computer loses all of its Narrow and Wide Range level inputs.

What will be the expected response to these failures?

a. Control will be transferred to the DCC-Y computer and a "Feedwater Computer (X) Trouble" alarm will be received.

b. DCC-X will revert to a level default value of +23 inches and maintain reactor water level.

c. All three Reactor Feedwater Pumps will lockup and a "Digital Feedwater Field Instrument Trouble" alarm will be received.

d. Reactor water level will rise rapidly to the Reactor Feedwater Pump and Main Turbine trip setpoints.

76 Unit 2 was operating at 100% power when it experienced a steam leak in the drywell equivalent to 60 gpm. The Unit Reactor Operator has maximized drywell cooling in accordance with OT-101, "High Drywell Pressure".

Which of the following describes the bases for maximizing drywell cooling?

Maximizing drywell cooling:

a. reduces the individual loads on all of the drywell fans.

b. is done to prevent receiving high drywell pressure isolation and scram signals.

c. is designed to make up for the heat addition from a failure of one Recirculation Pump seal.

d. provides additional time for steam leak location and isolation as the source of the pressure rise.

77 Given the following conditions on Unit 2:

- A leak in the drywell has resulted in rising temperature and pressure
- All plant systems responded as designed as pressure exceeded 2.0 psig
- T-102, "Primary Containment Control", was entered for high drywell temperature
- T-223, "Drywell Cooler Fan Bypass" has been completed

With the drywell cooling fans returned to service, what is the source of cooling water, if any?

a. Reactor Building Closed Cooling Water

b. Drywell Chilled Water

c. Emergency service water

d. No cooling water is available, the fans run for circulation only

78 A loss of coolant accident is in progress on Unit 3. The scram was successful and current conditions are as follows:

- Reactor water level
  -- Narrow Range (LI-94A, B & C) indicates +5 to +8 inches
  -- Wide Range LI-85A & B) indicates -10 inches
  -- Shutdown Range (LI-86) indicates +10 inches
  -- Fuel Zone instruments indicate -25 inches
- Drywell temperature (TI-2-501)
  -- Point 126 indicates 270 degrees F
  -- Point 127 indicates 267 degrees F
- Reactor Building temperature (TR-2-13-139)
  -- Point 22 indicates 155 degrees F
- Reactor pressure is 200 psig

Which of the following instruments is NOT AVAILABLE for reactor water level indication?

a. Narrow Range

b. Wide Range

c. Shutdown Range

d. Fuel Zone

TABLE DW/T-1

RPV LEVEL INSTRUMENT STATUS

AN RPV LEVEL INSTRUMENT MAY BE USED TO DETERMINE RPV LEVEL ONLY WHEN THE FOLLOWING CONDITIONS ARE SATISFIED:

NOTES:

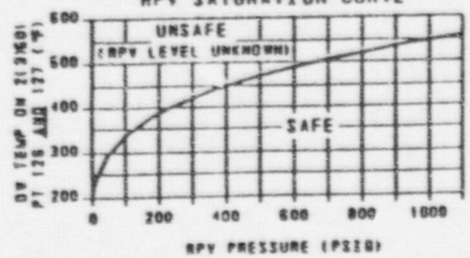
1. IF BOTH POINTS 126 AND 127 ARE AVAILABLE, THEN BOTH POINTS MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE

2. IF EITHER POINT 126 OR 127 IS NOT AVAILABLE, THEN THE REMAINING POINT MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE

ALL RPV LEVEL INSTRUMENTS

THE COMBINATION OF DW TEMP AND RPV PRESS IS IN THE SAFE REGION OF THE RPV SATURATION CURVE

RPV SATURATION CURVE



AND

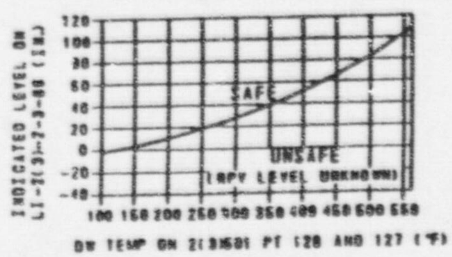
WIDE AND NARROW RANGE INSTS ONLY

FOR EACH OF THE INSTRUMENTS IN THE TABLE, THE INSTRUMENT READS ABOVE THE MIN INDICATED LEVEL OR THE TEMP NEAR THE DW REFERENCE LEG VERTICAL RUNS (TI-2(3)501 PT 126 AND 127) ARE BELOW THE MAX RUN TEMP.

INSTRUMENT	MIN INDICATED LEVEL IS ABOVE	OR	MAX RUN TEMP IS BELOW
NARROW RANGE	10 IN.	OR	450°F
WIDE RANGE	-120 IN	OR	600°F

SHUTDOWN RANGE INST LI-2(3)-2-3-86 ONLY

LI-2(3)-2-3-86 READS ON THE SAFE SIDE OF THE CURVE



79 Given the following conditions:

- Unit 2 is operating at 75% power
- One Safety Relief Valve has just opened and will not close
- Torus water temperature is 95 degrees F and rising

What are the actions required by OT-114, "Stuck Open Safety Relief Valve", for these conditions?

a. Reduce power in accordance with "GP-9-2, "Fast Reactor Power Reduction", then when Recirculation Pumps are at minimum, place the Mode Selector Switch in "Shutdown".

b. Place the Mode Selector Switch in "Shutdown" prior to torus temperature reaching 120 degrees F.

c. Scram the unit in accordance with GP-4, "Manual Reactor Scram".

d. Reduce power in accordance with GP-9-2, "Fast Reactor Power Reduction", then place the Mode Selector Switch in "Shutdown" when torus temperature reaches 110 degrees F

80 Given the following conditions:

- The System is under a Maximum Emergency Generation condition
- Unit 2 is raising power to 100%
- A loss of feedwater heating has occurred and power ascension was stopped at 90%
- Feedwater temperature is 20 degrees F below normal and lowering
- Reactor power is now trending upward
- PCIOMR surveillance is NOT required

What is the maximum power allowed by OT-104, "Positive Reactivity Addition", for these conditions?

a. 100%

b. 3112 MWt

c. 2766 MWt

d. 70%

81 Following a reactor startup on Unit 3, the Reactor Engineer reports that the plant has experienced a reactivity anomaly.

What specific parameters must be monitored to determine the magnitude of this anomaly?

a. Core thermal power and nuclear instrumentation indicated power

b. Total control rod worth and individual control rod worth

c. Actual control rod density and core thermal power

d. Predicted control rod density and actual control rod density

82 With Unit 3 at rated power the Mode Selector Switch is placed in "Shutdown" to scram the reactor. The full core display "Blue" scram lights are not lit.

These conditions directly indicate that:

a. the backup scram valves did not de-energize to vent the scram air header.

b. the "Overtravel Beyond Full-In" reed switches have not picked up for each control rod.

c. a scram discharge volume hydraulic lock has occurred.

d. an electrical failure of the RPS Trip Systems has occurred.

83 Unit 3 had been operating at full power for 412 days when a plant transient occurred requiring evacuation of the Main Control Room. Prior to leaving the Main Control Room all immediate actions directed by SE-1, "Plant Shutdown From The Remote Shutdown Panel", were completed.

What will be the expected approximate reactor pressure when the operators arrive at the remote shutdown panels? (Assume all systems are operating as designed.)

a. 940 psig

b. 1030 psig

c. 1135 psig

d. 1260 psig

84 Unit 2 is performing a plant cooldown in accordance with SE-10, "Alternative Shut Down". Reactor pressure is less than 75 psig. Cooldown is being accomplished using "Alternate Shutdown Cooling". How is the cooldown rate monitored while in this condition?

a. The local LPCI Pump discharge temperature.

b. The tailpipe temperature of the open Safety Relief Valve.

c. Reactor vessel skin temperatures at the main steam line level.

d. Main steam line saturated steam pressure converting to temperature per the SE-10 attachment.

85 With the Recirculation Pumps running at minimum speed, a loss of Reactor Building Closed Cooling Water occurs.

What are the restrictions on continued Recirculation Pump operation for these conditions?

In accordance with ON-113, "Loss of RBCCW", the Recirculation Pumps:

a. should be tripped immediately.

b. may continue to run as long as CRD purge is maintained on the seals.

c. should be tripped within one minute.

d. may continue to run as long as seal cavity temperatures remain within limits.

86 During a loss of instrument air the control rods begin to drift. Which of the following describes the direction of drift and what occurs to cause that movement?

The control rods drift:

a. in, similar to a normal insertion, because drive water flow and pressure both rise.

b. out, because a flowpath is opened to the top of the drive mechanism operating piston allowing reactor pressure to drift the rod out.

c. in, because the normal scram flowpath to and from the drive mechanism operating piston is opened allowing accumulator and reactor pressure to drift the rod in.

d. out, because a flowpath is opened from the bottom of the drive mechanism operating piston allowing accumulator pressure and gravity to drift the rod out.

87 Given the following conditions:

- Unit 2 is operating at 75% power
- All systems are operating as designed in automatic
- The Outboard MSIV in the "C" main steam line fails closed

Which of the following describes the expected plant response assuming no operator actions are taken?

a. Reactor pressure will spike, then stabilize at a value higher than its pre-transient value.

b. The remaining seven MSIVs close on high steam flow in the other steam lines.

c. Main generator output remains nearly constant and a MSIV closure half scram occurs.

d. Reactor pressure will spike, then return to its pre-transient value.

88 Given the following conditions:

- Unit 2 is in Mode 3
- Shutdown cooling is in service on the "A" RHR loop with a 8000 gpm flowrate
- Both Recirculation Pumps are shutdown
- Reactor water level by LI-86 (Shutdown Range) is out of service
- Reactor water level by LI-94 (Narrow Range) is 34 inches

In accordance with GP-12, "Core Cooling Procedure", the core is adequately cooled if: (Figure 1 of GP-12, "Core Cooling Procedure" is attached.)

a. present conditions are maintained.

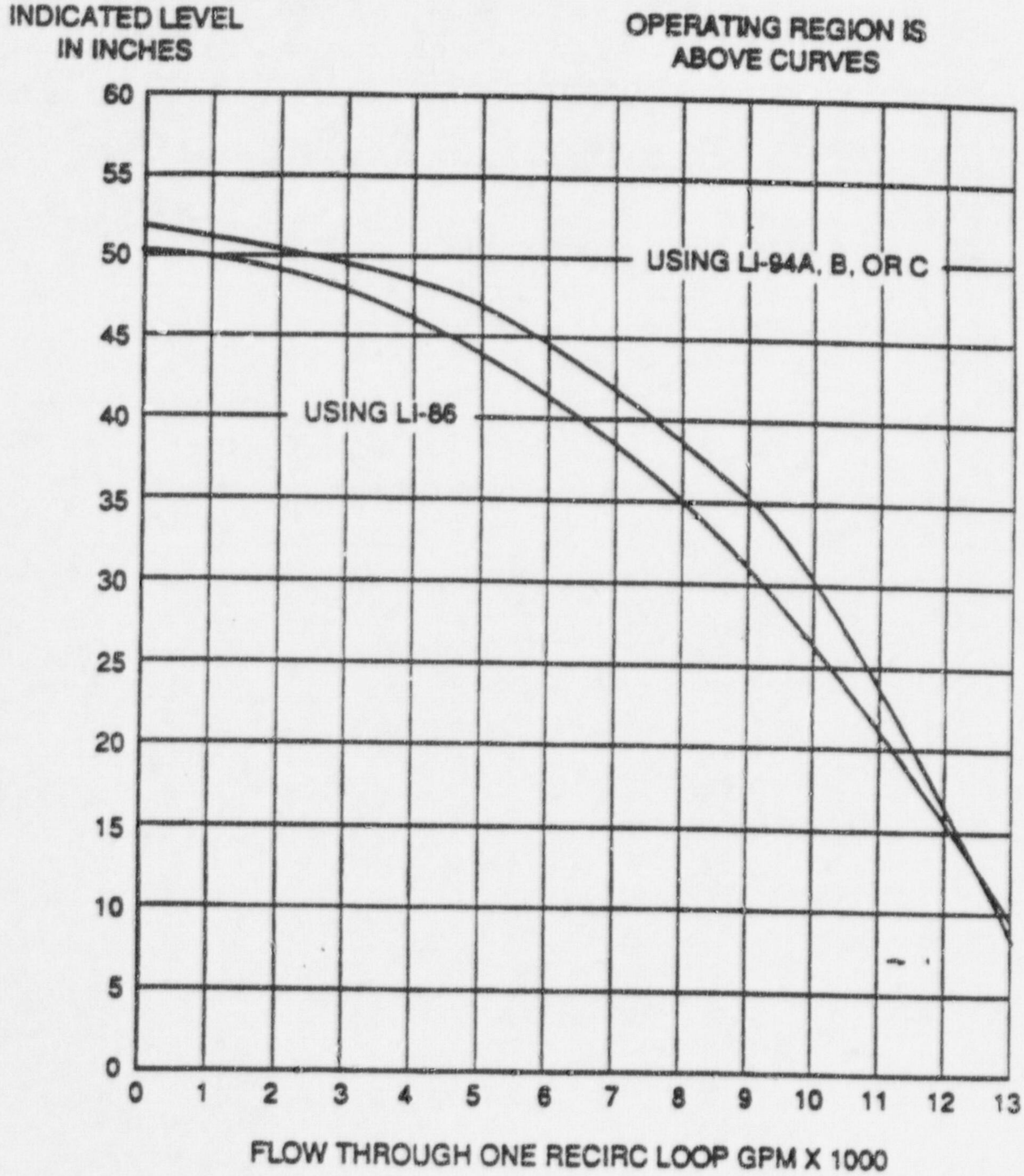
b. reactor water level is raised to 40 inches.

c. "A" RHR flow is raised to 8,500 gpm.

d. reactor water level is raised to 37 inches.

FIGURE 1

Vessel Level Vs. Flow Through One Recirc Loop



89 Given the following conditions:

- Unit 2 is shutdown in Mode 4
- Shutdown cooling is in service on the "A" Residual Heat Removal (RHR) loop using the "A" RHR Pump
- Panel 20Y33 has just been lost

Which of the following describes the core heat removal capabilities for these conditions?

a. Alternate decay heat removal systems should be utilized.

b. Shutdown cooling should be immediately transferred to the "B" RHR Loop.

c. The "A" RHR Pump should be reset and the "A" Loop returned to shutdown cooling

d. The RHR cross-tie valve should be opened to allow the "A" RHR Loop to return to the "B" Recirculation loop.

90 With Unit 2 operating at 90% power, annunciator 211 G-4, "CRD Charging Water Header High Pressure", is received.

Which of the following is the cause of this alarm and what is the expected impact on the plant?

a. The in-service CRD Flow Control Valve (AO-19) has failed closed resulting in rising Recirculation Pump seal temperatures.

b. The CRD Drive Water Pressure Control Valve (MO-20) has failed closed resulting in rising Recirculation Pump seal temperatures.

c. The in-service CRD Flow Control Valve (AO-19) has failed closed resulting in rising control rod drive mechanism temperatures.

d. The CRD Drive Water Pressure Control Valve (MO-20) has failed closed resulting in rising control rod drive cooling water flow.

91 Given the following conditions:

- Unit 2 is shutdown for refueling
- A Fuel Pool High Radiation alarm is received while moving a fuel assembly from the reactor vessel to its storage location
- The assembly was being lowered into a storage rack when the alarm occurred
- The assembly is NOT located near a radiation monitor

Which of the following describes what should be done with this fuel assembly prior to evacuation of the Fuel Floor?

The fuel assembly should immediately be:

a. returned to the reactor vessel and lowered back into its original location.

b. raised to above the top of the storage rack.

c. returned to the reactor vessel and left just above the upper grid.

d. lowered into its designated storage rack in the spent fuel pool

92 Which of the following MUST occur (and cannot be bypassed) to initiate Torus Sprays when required by T-102, "Primary Containment Control"?

a. A LPCI initiation signal must be present.

b. The Containment Spray Override 2/3 Core Coverage Switch (10A-S18) must be in "Override".

c. Reactor water level must be above -226 inches.

d. The Containment Spray Valve Switch (10A-S17) must be momentarily placed in "Manual".

93 With unit 2 operating at 100% power, one Safety Relief Valve (SRV) opened and did not reclose.

Which of the following tailpipe temperatures indicates the SRV is full open?

a. 212 degrees F

b. 250 degrees F

c. 315 degrees F

d. 544 degrees F

94 Which of the following is the reason why T-102, "Primary Containment Control", cautions against operating RCIC and HPCI with torus water temperatures above 190 degrees F?

a. The HPCI and RCIC Pump suctions will not transfer on low CST level.

b. The HPCI and RCIC turbine exhaust check valves will chatter.

c. The HPCI and RCIC Pumps will cavitate.

d. The torus will not condense all HPCI and RCIC turbine exhaust.

95 Given the following conditions:

- Unit 2 is operating at 100%
- One Safety Relief Valve has just opened and cannot be closed
- Drywell pressure and temperature are rising rapidly
- Drywell instrument run temperatures are rising rapidly
- Primary containment temperature control methods are unsuccessful

Assuming conditions continue to degrade, how would this impact the automatic initiation of the high pressure ECCS systems as reactor water level lowers?

High pressure ECCS systems:

a. initiate late because the wide range level indication will be reading 40 inches higher than actual level.

b. will not initiate because wide range level indication will be off-scale high.

c. initiate early because the wide range level indication will be reading 40 inches lower than actual level.

d. must be initiated by the operator because wide range level indication will be off-scale low.

96 Given the following conditions:

- Reactor pressure is 850 psig
- Torus water level is 17.5 feet
- Torus water temperature is 185 degrees F
- All controls rods are fully inserted

Which of the following would be expected if an emergency blowdown is performed for these conditions? See attached T-102 Curves.

a. Safety relief valves fail to operate.

b. Torus water will reach boiling sooner.

c. A reduction in Safety Relief Valve capacity (lbm/hour).

d. A direct discharge of steam to the containment.

97 Given the following conditions on Unit 3:

- A loss of coolant accident has occurred
- Torus water level is 11 feet
- Torus water temperature is 220 degrees F
- Torus pressure is 11.5 psig
- "A" RHR Loop flow is 9,500 gpm with the "A" Pump in operation
- "A" Core Spray loop flow is 3,500 gpm with the "A" Pump in operation
- The "B" RHR and "B" Core Spray Loops are not available
- Reactor water level has been recovered and is stable at +20 inches

Assuming reactor water level is maintained at it's current value, which of the following must be done? See attached T-102 ECCS Suction Requirement curves.

a. Reduce "A" Core Spray Pump flow to 2,500 gpm and raise "A" RHR Pump flow to 10,500 gpm.

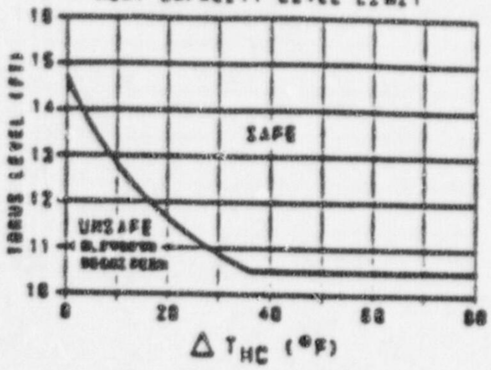
b. Start the "C" Core Spray Pump, throttle the "A" Core Spray Loop flow to 5000 gpm and throttle "A" RHR Loop flow to 8,000 gpm.

c. Secure the "A" Core Spray Pump and throttle the "A" RHR Pump flow to maintain 13,000 gpm.

d. Start the "C" Core Spray Pump, throttle the "A" Core Spray Loop flow to 7000 gpm and reduce "A" RHR Loop flow to 6000 gpm.

CURVE T/L-1

HEAT CAPACITY LEVEL LIMIT



ΔT_{HC} IN °F

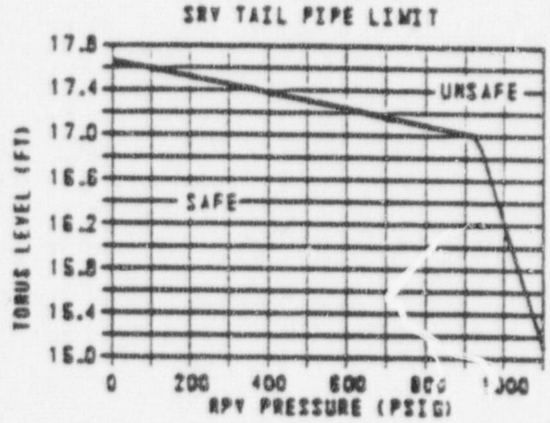
____ °F HEAT CAPACITY TEMP LIMIT FROM CURVE T/T-1

MINUS (-) ____ °F TORUS TEMP

EQUALS (=) ____ °F ΔT_{HC}

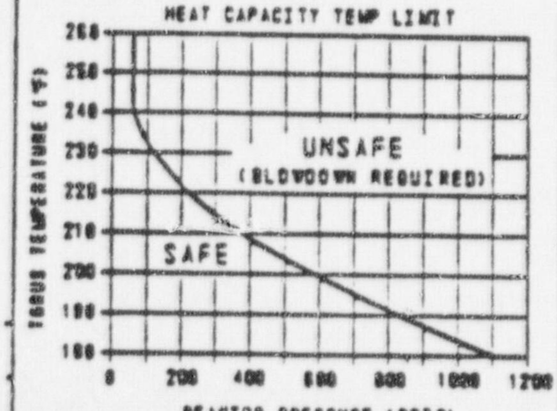
CURVE T/L-3

SRV TAIL PIPE LIMIT



CURVE T/T-1

HEAT CAPACITY TEMP LIMIT



98 Unit 3 has experienced a loss of feed transient. Which of the following describes the means of core heat removal once reactor water level has lowered to -210 inches?

a. Opening the Safety Relief Valves reduces fuel cladding temperature via heat transfer to the steam passing the fuel assemblies.

b. Allowing reactor water level to continue to lower provides steam cooling until the Minimum Alternate Flooding Pressure is reached.

c. Opening the Safety Relief Valves causes the remaining water in the vessel to swell to above the top of active fuel to provide cooling.

d. Allowing reactor water level to continue to lower provides steam cooling until the Minimum Steam Cooling RPV Water Level is reached.

99 Given the following conditions:

- Unit 2 has entered T-103, "Secondary Containment Control" after receiving an alarm for high water level in the HPCI Room
- Condensate Storage Tank water level is dropping slowly
- No abnormal HPCI room equipment operation is apparent

In accordance with T-103, "Secondary Containment Control", which of the following is an acceptable means of determining that water level is at or above the "Action" level in the room?

a. Direct observation of the placard in the HPCI Room by opening the door from the RCIC Room.

b. Reactor Building Floor Drain Sump Hi-Hi Level alarm is received.

c. Direct observation of the water level in the stairwell outside the HPCI Room.

d. The HPCI Room water level computer point on SPDS is reading above 2 feet.

100 Given the following conditions:

- Unit 3 had a scram condition while at 100% power
- Several control rods failed to insert on the scram signal
- Reactor pressure is being controlled by the main turbine bypass valves
- No determination on the cause of the failure to scram has been made

Which of the following actions will result in motive force being applied to scram the rods WITHOUT REGARD for the cause of the failure?

a. De-energize the scram solenoids.

b. Isolate and vent the scram air header.

c. Vent the control rod drive overpiston area.

d. Reset the scram, drain the scram discharge volume and initiate a manual scram.

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

Applicant Information

Name:	Region: I
Date: 09/11/98	Facility: Peach Bottom
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 four 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

PART 1

PECO NUCLEAR

STATION PB / LGS

COURSE TITLE LOT 9701 NRC EXAMS

FORM SRO

ANSWER KEY

NAME ______ PRINT last ______ first ______ mi ______

SOCIAL SECURITY NUMBER ______

COMPANY / PECO PAYROLL # ______

DATE ______

I HAVE REVIEWED AND UNDERSTAND THE CORRECTED QUIZ; ALL WORK ON THIS EXAMINATION IS MY OWN. I HAVE NEITHER GIVEN NOR RECEIVED ASSISTANCE ______ signature

IMPORTANT

- USE #2 PENCIL
- EXAMPLE: A B ● D E
- ERASE *COMPLETELY* TO CHANGE

Question	Answer	Question	Answer
1	B (T / F – KEY)	51	A
2	A – DELETED	52	C
3	D	53	D
4	B	54	C
5	A	55	B
6	A	56	C
7	D	57	B
8	C	58	A
9	C	59	D
10	A	60	C
11	D	61	D
12	B	62	C
13	A	63	A
14	C (#14 NOTE: THIS SHOULD HAVE [illegible])	64	C
15	D	65	B
16	B	66	C
17	B	67	B
18	D	68	D
19	C	69	A
20	B	70	D
21	B	71	B
22	C – DELETED	72	C
23	D – DELETED	73	C
24	A	74	D
25	B	75	D
26	C	76	B
27	D	77	C
28	A	78	B
29	D	79	A
30	C	80	D
31	B	81	C
32	B	82	A
33	C	83	B
34	D	84	A
35	A	85	C
36	B	86	B
37	A	87	A
38	A	88	D
39	B	89	C
40	D	90	C
41	B	91	A
42	A	92	D
43	D	93	B
44	C	94	D
45	A	95	B
46	D	96	A
47	C	97	A
48	B	98	C
49	B	99	C
50	D	100	D

50

50

A C

Senior Reactor Operator Examination

1 Given the following conditions:

- A Reactor Operator (RO) has just returned to shift following 2 weeks vacation
- The RO's license is active and current
- On the first day back on shift, this RO worked a normal 12-hour shift and then accepted and worked 2 hours of overtime to cover for an absent operator

What is the maximum number of hours this RO may work on the next day (his second day back on shift)? Assume no additional authorizations have been made.

a. 8 hours

b. 10 hours

c. 12 hours

d. 14 hours

2 The Unit 2 Unit Reactor Operator (URO) is being temporary relieved from his duties at the control boards for a break.

The individual providing the temporary relief shall:

a. walkdown all Unit 2 control panels and shall have attended the Shift Turnover Meeting.

b. perform a complete turnover in accordance with OM-C-6.1, "Shift Turnover" and shall NOT hold the URO position for more than 60 minutes.

c. perform a complete turnover in accordance with OM-C-6.1, "Shift Turnover", and have attended the Shift Turnover Meeting.

d. walkdown all Unit 2 control panels and shall NOT hold the URO position for more than 30 minutes.

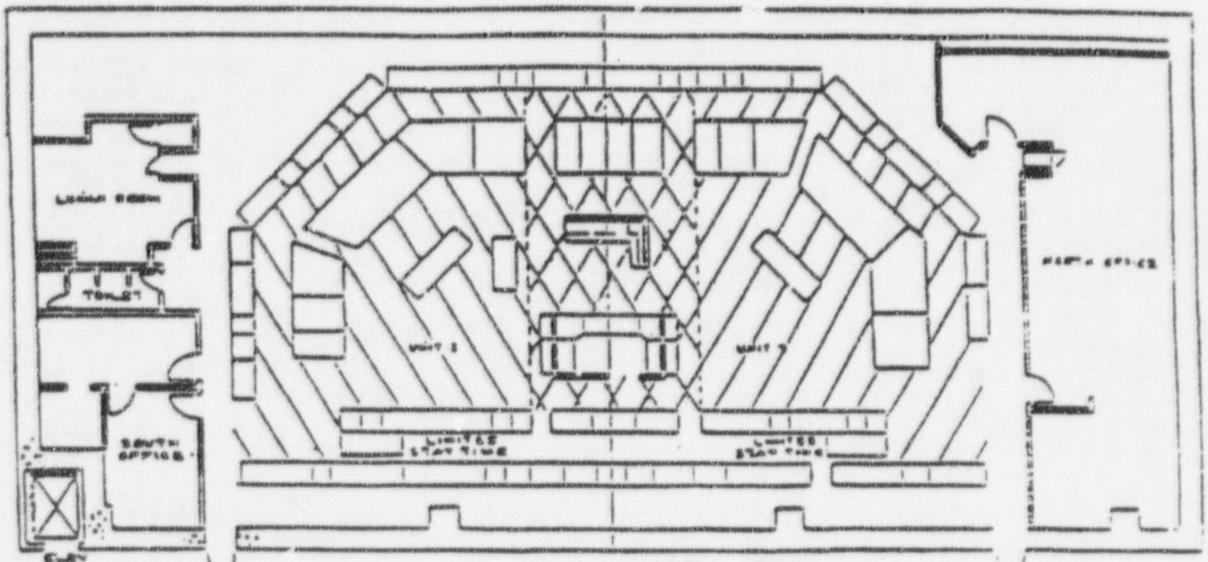
3 In accordance with OM-P-3.2, "Senior Licensed Operators", select the conditions that would allow the Control Room Supervisor (CRS) to go to the recombiner panel to supervise the Plant Reactor Operator placing the standby jet compressor in service. Refer to the attached Main Control Room sketch.

a. The Work Control Supervisor is in his office in the Control Room.

b. A "temporary relief" turnover is required to be completed.

c. The Unit 2 Unit Reactor Operator is able to maintain the CRS in sight at all times.

d. The Shift Manager remains at the Control Room Supervisor desk.



MAIN CONTROL ROOM

4 Given the following conditions:

- Unit 3 is performing a shutdown with control rod insertions in progress
- Reactor power is 16% with generator output at 180 MWe
- The Plant Reactor Operator is deinerting the drywell
- The CRS is monitoring the shutdown from the CRS desk

Which of the following additional requirements, if met, would allow a License Class Instant SRO trainee under direction of the shutdown Reactor Operator to continue rod motion for the given conditions?

a. The Reactor Engineer is present to satisfy Technical Specification requirements.

b. The 4th Reactor Operator is performing Double Verification for control rod movements.

c. The Shift Manager has conducted a shift briefing on control rod insertions.

d. The Vice President-PBAPS written permission has been received allowing trainees to manipulate control rods.

5 A clearance is being prepared on Unit 3. The system being tagged does not have controlled drawings available.

The clearance shall be prepared:

a. following a system walkdown.

b. and approved by the Work Week Manager.

c. and then a technical review is required by the System Manager.

d. using the vendor technical manuals.

6 Which of the following would REQUIRE a second individual to actually witness the activity while it is occurring?

a. Restoration of a throttled valve to its required locked position.

b. Fuse removal as directed by the T-200 procedures.

c. Restoration of a clearance on a ECCS System.

d. A routine surveillance test being performed in a Radiation Area.

7 Given the following conditions:

- Unit 2 has been operating at 100% power for several months
- At 1900 on 09/14/98 it is discovered that due to a recent procedure change, part of a TS required surveillance was not performed
- The last surveillance was satisfactorily completed at 1900 on 09/01/98
- The incomplete surveillance was performed at 1900 on 09/13/98
- The surveillance is required to be performed at least once per 12 days
- The action statement for this equipment requires that it be restored within 72 hours, or be in Hot Shutdown within 12 hrs and in Cold Shutdown within the next 24 hours

If the remaining portion of the surveillance is not able to be performed, when is the Unit required to be in Hot Shutdown?

a. By 1900 on 09/17/98

b. By 0700 on 09/18/98

c. By 1900 on 09/19/98

d. By 0700 on 09/20/98

8 Given the following conditions:

- Plant Systems "A" and "B" are Tech Spec systems required to support the operation of Tech Spec System "C"
- If inoperable, System "A" has a completion time for restoration to Operability of 7 days
- If inoperable, System "B" has a completion time for restoration to Operability of 14 days
- If inoperable, System "C" has a completion time for restoration to Operability of 3 days
- Today, System "A" has been determined to be Inoperable

Which of the following is the MAXIMUM time (from today) for restoring the System "C" to an Operable Status?

a. 3 days

b. 7 days

c. 10 days

d. 17 days

9 Given the following conditions:

- A male, fully qualified Peach Bottom radiation worker is scheduled to work in a Level I - Locked High Radiation Area (Assume area radiation levels are at the minimum for the area classification)
- Total Effective Dose Equivalent TEDE exposure for 1998 for this worker is 985 mrem
- This is a non-emergency situation with no exposure limit extensions authorized

Without exceeding any exposure limits, this worker may remain in this area for a maximum of: (Choices are rounded down to the nearest minute.)

a. 12 minutes

b. 18 minutes

c. 120 minutes

d. 180 minutes

10 Given the following conditions:

- A scheduled Unit 2 surveillance is required to be performed on a system in a radiation area
- All radiological precautions have been taken and a pre-evolution brief has been completed

Using the As Low As Reasonably Achievable (ALARA) guidelines, which of the following is the PREFERRED method for completing this surveillance? (Consider only the personnel aspects of this surveillance.)

a. One individual installing shielding in a 90 mr/hour area for 30 minutes then performing the surveillance in a 9 mr/hour area for 60 minutes.

b. Two individuals performing the surveillance in a 90 mr/hour area for 35 minutes.

c. One individual performing the surveillance in a 90 mr/hour area for 60 minutes.

d. Two individuals installing shielding in a 90 mr/hour area for 15 minutes then performing the surveillance in a 9 mr/hour for 35 minutes.

11 During a declared emergency, it is necessary to raise the PECO Administrative Dose Control Levels to the NRC annual exposure limits. Which of the following describes how this extension is authorized?

a. During a declared emergency, all Peach Bottom qualified Radiation Workers are automatically extended to the NRC TEDE limit.

b. The Radiation Protection Manager provides the Emergency Director with verbal case-by-case extension authorizations to the NRC limit

c. The Control Room Supervisor provides immediate verbal extension authorization for Operations personnel.

d. The Emergency Director approves a "Dose Extension Form".

12 A check-off list (COL) Independent Verification (IV) is required to be completed on 8 system valves located in an area with dose rates of 120 mr/hour.

What is the maximum time available to complete the verification before exceeding the guidelines for Shift Management to consider waiving the IV?

a. 2 minutes

b. 5 minutes

c. 10 minutes

d. 12 minutes

13 A Main Control Room annunciator has a "blue" dot on its window.

Which of the following describes the status of the equipment monitored by that annunciator?

The monitored equipment has a deficiency that:

a. affects the performance of the Transient Response Implementation Plan (TRIP) procedures.

b. is not considered a Main Control Room deficiency.

c. affects the performance of the Emergency Response Procedures (ERP).

d. does not impact any safety related plant equipment.

14 The Shift Manager (SM) is reviewing the final report and computer printouts on 09/15/98 for a Technical Specification required shutdown that occurred on Unit 2 on 09/13/98. He determines that the Unit did not reach Hot Shutdown within the required Tech Spec time limits thus exceeding the Emergency Action Level for an Unusual Event. No emergency declarations were made on 09/13/98.

Select the required action(s) for these conditions.

a. The event should be classified and reported but not declared.

b. The NRC Resident Inspector shall be notified within one hour.

c. An Unusual Event should be declared.

d. Classifications and notifications are NOT required, but the missed Unusual Event shall be included in the Licensee Event Report (LER).

15 The Plant Reactor Operator (PRO) has just received a fire alarm from the Turbine Building.

The PRO is REQUIRED to make a call for off-site fire fighting support:

a. after 10 minutes if an actual fire is confirmed.

b. immediately if equipment for safe shutdown is jeopardized.

c. when 2 or more fire alarms are received in the same area.

d. after 20 minutes if the Incident Commander reports the fire is NOT controlled.

16 Given the following conditions:

- A plant transient occurred on Unit 3 at 1415 resulting in an Unusual Event declaration at 1425
- While completing the Unusual Event Notification Form an Alert was declared at 1435 for an unrelated event

The State and Local Agencies shall be notified of the Alert no later than:

a. 1440.

b. 1450.

c. 1515.

d. 1535.

17 A LOCA occurred on Unit 3 at 0700. Plant conditions deteriorated requiring a General Emergency (GE) to be declared at 0715.

Which of the following is the LATEST time that the Emergency Director shall issue a Protective Action Recommendation (PAR)?

a. 0715

b. 0730

c. 0800

d. 0815

18 Given the following conditions:

- Unit 2 is making preparations for a reactor startup from a refueling outage
- Reactor Building ambient temperature is 74 °F
- The Reactor Building Equipment Operator is charging the hydraulic control unit accumulators with nitrogen to a pressure of 590 psig
- Several days later with the Unit at 100% power, Reactor Building temperatures have stabilized at 92 °F

Which of the following describes the expected impact on the Control Rod Drive Hydraulic system operations for these conditions? (Refer to attached figure.)

The individual control rod:

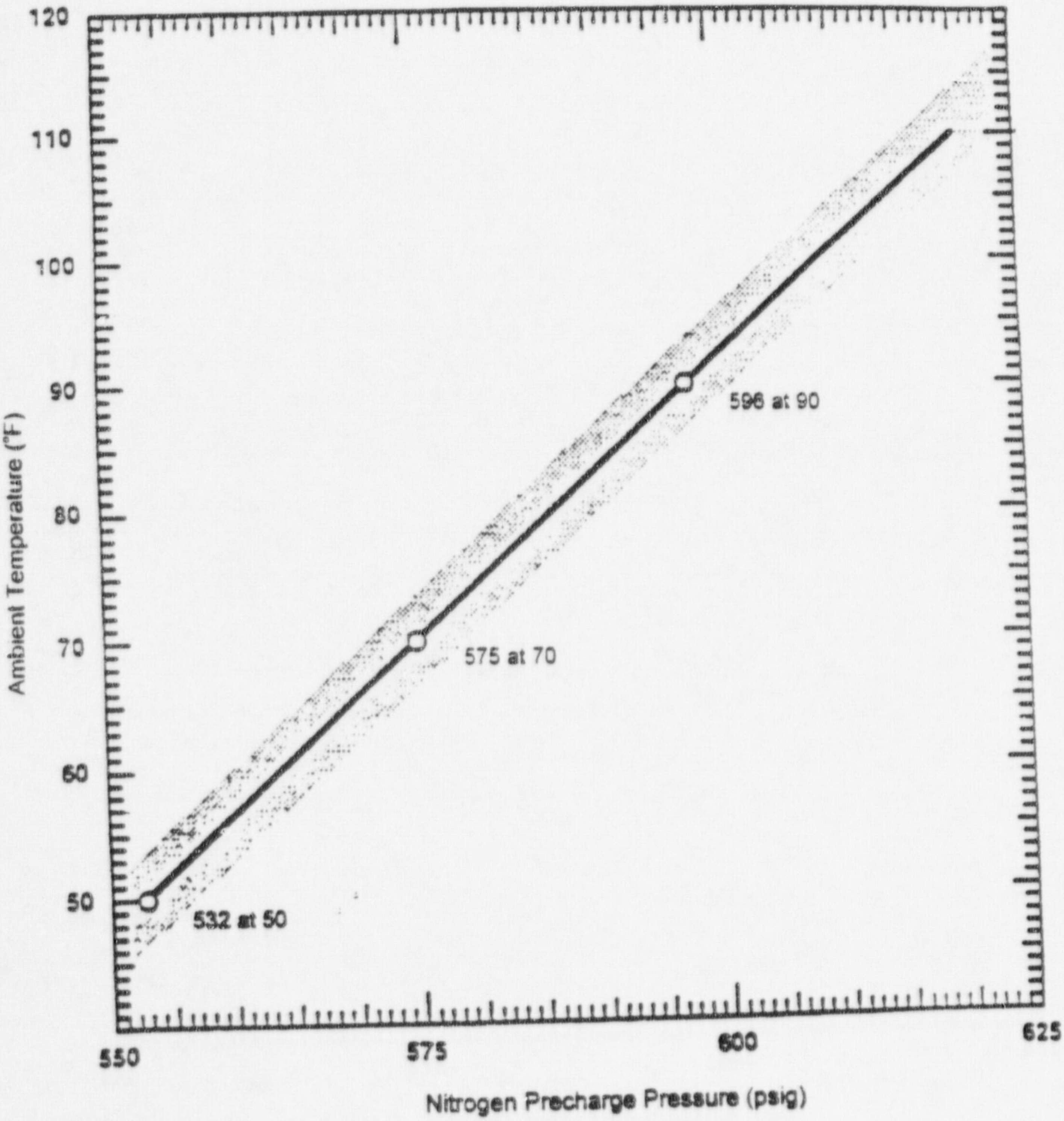
a. normal insertion speeds will be slower and may result in control rod drift alarms.

b. scram speeds will be slower and will result in reduced reactivity addition rates.

c. normal insertion speeds will be faster and may result in "double notching".

d. scram speeds will be faster and may result in mechanism damage.

ACCUMULATOR PRECHARGE NITROGEN PRESSURE VERSES AMBIENT TEMERATURE



19 A loss of the "B" Reactor Protection System (RPS) MG Set has just occurred on Unit 3.

Which of the following describes how this will impact the Backup Scram Valves (SV-140A and 140B)?

Loss of the RPS Bus will:

a. not affect the DC power to the Backup Scram Valves and they will remain energized

b. cause a loss of AC power opening two Backup Scram Valves such that a loss of the other RPS Bus will result in a full scram.

c. not affect the DC power to the Backup Scram Valves and they will remain de-energized.

d. cause a loss of AC power opening one of the two Backup Scram Valves required to vent the scram air header.

20 Given the following conditions:

- Control rod 34-47 has just been withdrawn one notch (Notch "44" to "46")
- The Rod Control Switch was used for the withdrawal in the "Rod Out Notch" position
- The Reactor Manual Control System (RMCS) operated as designed during the rod withdrawal except the amber "settle" light did not extinguish when expected

What is the expected status of the Control Rod Drive Hydraulic System and control rod 34-47 for this failure?

a. The Reactor Manual Control System Master Timer did not time out and the Auxiliary Timer is controlling rod movement and will settle the rod at Notch "46".

b. The hydraulic control unit withdrawal exhaust directional control valve has not closed and the control rod has settled at Notch "46".

c. The Reactor Manual Control System Master Timer did not time out and the Auxiliary Timer is allowing continued rod withdrawal until it times out.

d. The hydraulic control unit withdrawal exhaust directional control valve has not closed and the control rod will drift full out.

21 Given the following conditions:

- Unit 2 was operating at 75% power with a full power rod pattern
- A loss of Main Generator Stator Cooling occurred and was reset in 5 seconds
- The Unit Reactor Operator identified indications of Thermal Hydraulic Instabilities (THI) and scrammed the reactor
- Reactor pressure dropped to 900 psig and has stabilized at 950 psig
- Reactor water level dropped to -38 inches and has been stabilized at +20 inches

What is the status of the Reactor Recirculation System?

a. Both Recirculation Pumps are tripped.

b. The "A" Recirculation Pump is tripped, the "B" Pump is running at 30% speed.

c. Both Recirculation Pumps are running at 30% speed.

d. The "B" Recirculation Pump is tripped, the "A" Pump is running at 30% speed.

22 The Unit 2 Reactor Operator (URO) is starting the "A" Recirculation Pump. After the MG Set Drive Motor Start/Stop switch is placed in "Start" a scoop tube lock occurs. Generator speed when the scoop tube lock occurred was 45% (and lowering to 30% as designed). Assume no additional operator actions.

Which of the following describes the impact on the Recirculation Pump Motor Generator Set for these conditions?

a. The MG Set Drive Motor Breaker will open on exciter field overcurrent.

b. An MG Set lockout will occur on Incomplete Sequence.

c. The MG Set Drive Motor Breaker will open on motor feeder overcurrent.

d. An MG Set lockout will occur on Scoop Tube Lock

23 Given the following conditions:

- Unit 2 has just completed a power change from 75% to 100% using recirc flow
- Over the next hour, the operators note that power is continuing to slowly rise
- No operator actions are being taken

Which of the following would be the cause of this power rise?

a. Xenon concentration is rising from the power change.

b. Feedwater temperature are rising following the 25% power change.

c. Core inlet subcooling is greater at the higher recirculation flows.

d. Recirculation Pump MG Set oil temperatures are rising.

24 Given the following conditions:

- A power change from 75% to 80% is in progress on Unit 3
- The Unit Reactor Operator (URO) is raising speed on the "B" Recirculation Pump
- While pump speed is rising, the "B" Recirc Pump Speed Control Signal fails downscale

The "B" Recirculation Pump speed change will:

a. stop and local control of the scoop tube will be required to complete the power change.

b. stop, the pump will runback to 30% speed and the scoop tube will lock up.

c. continue until the operator trips or manually locks up the pump.

d. continue until the pump reaches the highest speed set by the URO, then the scoop tube locks up.

25 With Unit 2 and Unit 3 at power, a loss of DC power to the Unit 2 RHR System I logic has occurred. While troubleshooting is in progress, a valid LOCA signal on Unit 2 occurs.

How does this failure and initiation signal impact the Unit 2 and Unit 3 Residual Heat Removal (RHR) System operation?

a. The Unit 2 "B" RHR loop will automatically initiate in the LPCI mode, the Unit 2 "A" RHR loop must be started manually and any running Unit 3 RHR pumps will trip.

b. Both Unit 2 RHR loops will have a normal automatic initiation in the LPCI mode and any running Unit 3 RHR Pumps will automatically trip.

c. The Unit 2 "B" RHR loop will automatically initiate in the LPCI mode, the Unit 2 "A" RHR loop must be started manually and any running Unit 3 RHR pumps must be stopped by the operator.

d. Both Unit 2 RHR loops will have a normal automatic initiation in the LPCI mode and any running Unit 3 RHR Pumps must be stopped by the operator.

26 ST-0-010-306-2, "B" RHR Loop/Pump/Valve/Flow/And Unit Cooler Functional And Inservice Test, is being performed on Unit 2. During the first step of procedure, the "B" Torus Suction Valve (MO-013B) was stroke timed closed. While in this alignment, a valid LOCA signal occurred. The following conditions exist:

- Reactor pressure is 280 psig and lowering slowly
- The "B" RHR Loop Inboard Injection Valve (MO-25B) has opened
- The "D" RHR Pump started and then tripped on overload
- All other systems are operating as designed and no operator actions have been taken

Which of the following describes the "B" LPCI Loop status?

a. The loop is NOT injecting because reactor pressure is above the shutoff head of the running "B" RHR Pump.

b. The loop is injecting after automatically realigning for LPCI injection.

c. The loop is NOT injecting because the "B" RHR Pump is not running.

d. The loop is injecting at 5500 gpm due to current reactor pressure.

27 Given the following conditions:

- Unit 3 has had a complete loss of the E13 4160VAC Bus
- This results in a loss of power to the "A" Residual Heat Removal (RHR) Pump and to the "A" Loop Inboard LPCI Injection Valve (MO-25A)
- A valid LOCA signal occurs

What must occur to result in a final, design injection flowrate for these conditions of 30,000 gpm.

a. The RHR Loop Cross-Tie Valve (MO-20) must be unlocked and opened by an operator.

b. An operator must manually transfer the Inboard LPCI Injection Valve (MO-25A) to the alternate power supply.

c. The Outboard LPCI Injection Valve (MO-154A) must automatically open to inject though the normally open MO-25A.

d. The Inboard LPCI Injection Valve (MO-25A) must automatically transfer to the alternate power supply.

28 Given the following conditions:

- Unit 2 is shutdown with fuel handling operations in progress
- The "A" Loop of Residual Heat Removal (RHR) is operating in the Shutdown Cooling Mode
- A leak occurs between the Shutdown Cooling Suction Valves (MO-17 and 18) rapidly lowering reactor water level and resulting in a LPCI initiation signal on Lo-Lo-Lo level
- All expected actions occur

What is the expected response of the Shutdown Cooling System and reactor water level for these conditions? (Do not consider Core Spray in your selection of answers.)

a. Shutdown cooling will isolate. Reactor level will stabilize but not recover unless operator action is taken to inject.

b. Shutdown cooling will NOT automatically isolate. Operator action is required to isolate the leak and inject with RHR to recover level.

c. Shutdown cooling will isolate. Reactor level will rise due to the "B" Loop of RHR injecting in the LPCI mode until stopped by the operator.

d. Shutdown cooling will NOT automatically isolate. Operator action is required to isolate the leak allowing automatic LPCI injection to recover level.

29 Given the following conditions:

- The Unit 2 High Pressure Coolant Injection (HPCI) system was manually initiated using the SO procedure due to lowering reactor water level
- The HPCI Manual Initiation Pushbutton was used
- The HPCI Manual Initiation Pushbutton stuck in the "Armed" position
- Water level has been returned to the normal band and is stable

Which of the following is the correct method to shutdown HPCI in accordance with SO 23.2.A-2, "HPCI System Shutdown"?

a. Depress and hold the HPCI System Remote Trip Pushbutton until turbine speed indicates "0" rpm, then place the Aux Oil Pump in "Pull-To-Lock".

b. Depress the Local Manual Trip on the HPCI Turbine pedestal.

c. Depress and hold the HPCI System Remote Trip Pushbutton, close the Steam Admission Valve (MO-14) and then release the pushbutton.

d. Depress the "HPCI Isolation" pushbutton.

30 With the High Pressure Coolant Injection (HPCI) operating and injecting to the reactor vessel a loss 480 VAC power supplying HPCI components occurs.

What is the expected result of this failure?

a. Loss of speed control resulting in a trip of HPCI on overspeed

b. Inability to remove gland seal non-condensables causing high airborne conditions in the HPCI Room.

c. Inability to isolate a leak on the HPCI steam line immediately outside the drywell.

d. Loss of the Auxiliary Oil Pump on HPCI shutdown resulting in damage to the HPCI pump and turbine bearings.

31 Given the following conditions:

- The Unit 2 "System II Core Spray to Top Of Core Plate High Diff Pressure" alarm has been received
- Reactor pressure is 300 psig
- Drywell pressure is 0.35 psig and steady

With the "B" and "D" Core Spray Pumps running, system flow will be going into:

a. the area inside the core shroud on top of the core but bypassing the Core Spray spa.ger.

b. the downcomer area between the core shroud and the reactor vessel wall.

c. the drywell area between the maintenance valve and the reactor vessel wall.

d. the drywell area between the Core Spray maintenance valve and the testable check valve.

32 In order for Peach Bottom to meet the "ATWS Rule", 10CFR50.62, requirement for an equivalent Standby Liquid Control System (SBLC) flowrate of 86 gpm of 13% (by weight) sodium pentaborate for injection during an ATWS:

a. the contents of the storage tank must be maintained above 73 degrees F to meet concentration requirements.

b. Boron-10 concentration must be enriched to meet injection requirements.

c. the entire contents of the storage tank must be injected to meet concentration requirements in the reactor.

d. the capacity of the SBLC Pumps was raised to meet this requirement.

33 Given the following conditions:

- Unit 3 experienced a Group I isolation
- The reactor scrammed and all normal scram action: ere completed
- The Scram Discharge Volume Keylock Switch has been placed in "Bypass"

Which of the following would prevent resetting the scram?

a. Condenser vacuum is 18 inches.

b. The Main Steam Isolation Valves are all closed.

c. Reactor level is "0" inches.

d. Reactor pressure is 1075 psig.

34 Unit 2 is operating at full power. While transferring the "A" Reactor Protection System (RPS) Bus from the Normal to the Alternate power supply on Unit 2, the Transfer Switch malfunctions and "Alternate" power is unable to be selected. It is decided to leave the bus deenergized for initial troubleshooting of the switch.

Which of the following is an RPS consideration with this bus deenergized for a long time period?

a. Continuous operation with one of the Backup Scram Valves open.

b. A possible loss of redundancy with the "one-out-two-taken-twice" logic bypassed.

c. Continuous operation with two of the four ARI valves open.

d. A possible control rod block and eventual reactor scram on Scram Discharge Volume high levels.

35 With Unit 2 at 100% a transient requiring control rod insertions per GP-9-2, "Fast Reactor Power Reduction", occurs. While in progress the "RPIS Inoperative" alarm is received. The Unit Reactor Operator stops driving rods to investigate the annunciator.

Continued control rod insertions are possible:

a. only by placing the Mode Selector Switch in "Shutdown".

b. for all control rods with the exception of the rod with the RPIS failure.

c. by using the Emergency In/Notch Override Switch in "Emergency In".

d. by using the Rod Control Switch held in the "In" position.

36 A Traversing Incore Probe (TIP) trace is in progress on Unit 2 when a high drywell pressure occurs. After checking the TIP Valve Control Monitor the Plant Reactor Operator reports the following indications (Refer to attached figure):

- "Squib Monitor" lights - both extinguished
- "Shear Valve Monitor" lights - both extinguished
- "Ball Valve Open" lights - both illuminated
- "Ball Valve Closed" lights - both extinguished

Which of the following describes the status of the TIP system and the required operator actions?

a. The system has responded as designed. Operator action is required to close the ball valves.

b. The TIP detectors may not have withdrawn. Withdraw the detectors and verify the ball valves close.

c. The system has responded as designed. Operator action is required to fire the shear valves.

d. The TIP detectors may not have withdrawn. Fire the shear valves, withdraw the remaining cable and close the ball valves.

37 Given the following conditions on Unit 3:

- A startup is in progress with power at 4%
- The Mode Selector Switch (MSS) is in "Startup"
- Reactor pressure is 1000 psig
- Reactor water level is +23 inches
- No APRMs/WRNMs are bypassed

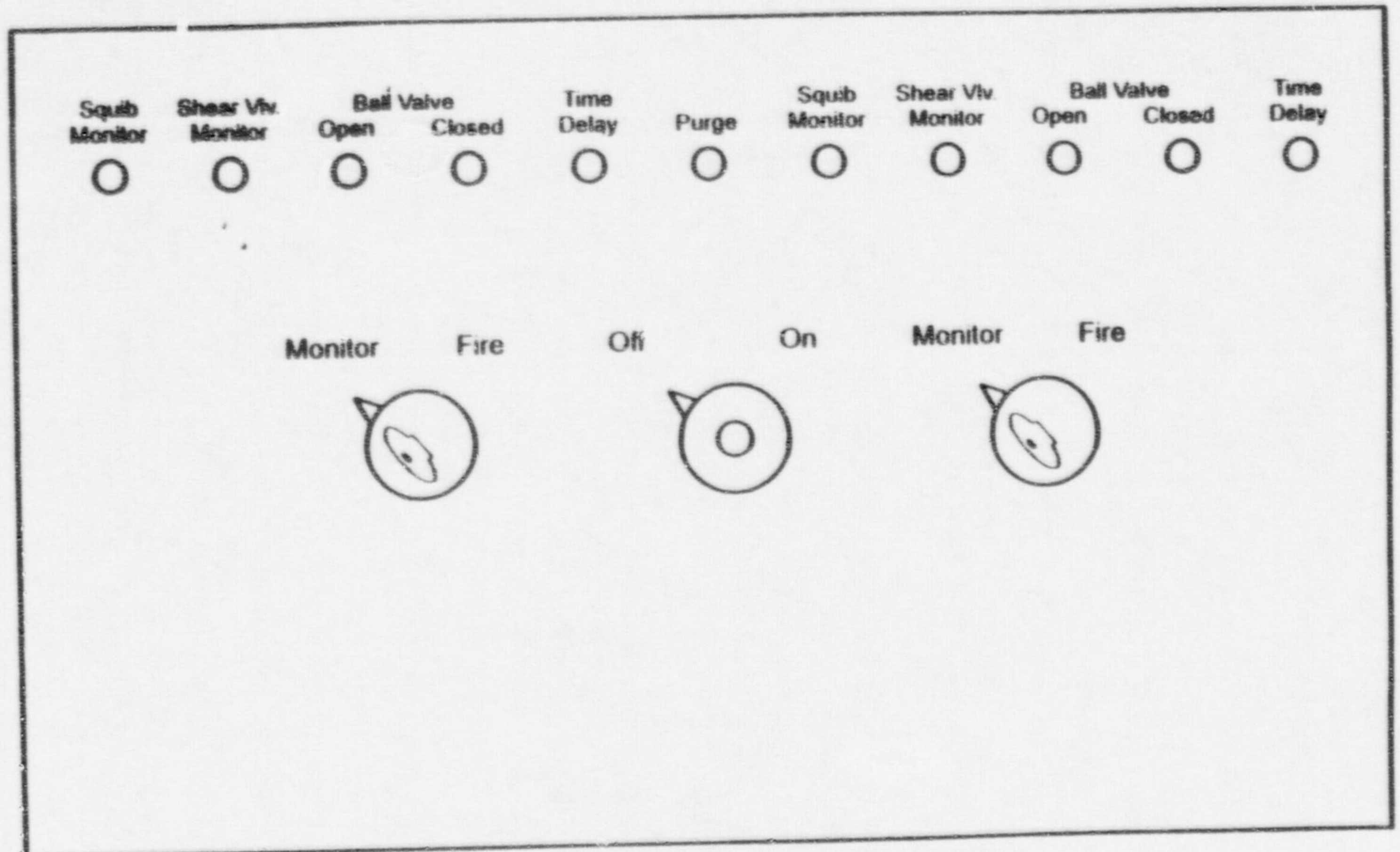
How will a loss of both 24 VDC power supplies to the nuclear instrumentation system impact the continuing startup?

a. An immediate reactor scram will occur.

b. An immediate APRM control rod block will occur.

c. A reactor scram will occur if the plant reaches 12% power without placing the MSS in "Run".

d. An APRM control rod block will occur when the MSS is placed in "Run" at 5% power.



TIP VALVE CONTROL MONITOR

38 Given the following conditions:

- A failure-to-scram (ATWS) has occurred on Unit 2
- Reactor power 22% and steady
- Reactor water level +10 inches and steady
- Reactor pressure 1010 psig and steady
- Drywell pressure 1.7 psig and rising slowly
- Scram Discharge Volume Vent and Drain Valves are closed
- Scram Discharge Volume High Level Scram is in
- Full core display blue scram lights are all illuminated
- Mode Selector Switch is in "Shutdown"

What actions must be taken before the scram can be reset to allow draining the Scram Discharge Volume for these conditions?

a. The Reactor Protection Trips must be bypassed using jumpers.

b. The SDV High Level Scram Bypass Switch must be placed in "Bypass"

c. The Mode Selector Switch must be taken out of "Shutdown" and the SDV High Level Scram Bypass Switch placed in "Bypass"

d. An immediate scram reset is possible after the Mode Selector Switch in "Shutdown" scram signal time delay has expired.

39 Given the following conditions:

- A Unit 2 startup is in progress with power at 20%
- Recirculation flow is 30%
- The "A" APRM Flow Unit output remains at 30% as recirculation flow is raised

As the plant startup continues, what will be the FIRST protective action to occur and the reason for that action?

a. A control rod block will occur due to flow biased neutron flux upscale.

b. A control rod block will occur due to a flow unit comparator trip.

c. A half scram will occur due to a flow unit "inop" signal.

d. A full scram will occur due to flow biased neutron flux upscale.

40 Unit 2 is performing a startup and heatup from cold shutdown conditions. As the plant heats up the Unit Reactor Operator reports that the wide range indicated level is continuing to go down at a steady rate. All plant systems are operating as designed and reactor water level is normal and steady.

Which of the following describes the cause of this level indication problem?

a. The density compensation signal (reactor pressure) has failed full "upscale".

b. The wide range level transmitter reference leg instrument isolation valve is closed.

c. The recirculation flow rise results in an indicated level rise.

d. The wide range level transmitter variable leg instrument isolation valve is closed.

41 Given the following conditions:

- Unit 2 has experienced a loss of all AC power (station blackout)
- The Reactor Core Isolation Cooling (RCIC) system automatically initiated
- Reactor water level is now -52 inches and rising
- The Control Room Supervisor directs the Unit Reactor Operator to isolate RCIC

What will be the expected RCIC system response when the operator depresses the Manual Isolation Pushbutton?

a. A normal RCIC system isolation and turbine trip will occur.

b. A RCIC turbine trip and system isolation will occur except the Inboard Steam Isolation Valve (MO-15) will not close.

c. No RCIC isolation actions or turbine trip will occur.

d. A RCIC turbine trip and system isolation will occur except the Outboard Steam Isolation Valve (MO-16) will not close.

42 Unit 2 requires an Emergency Blowdown after performing steam cooling in T-111, "Level Restoration". All actions required by T-112, "Emergency Blowdown", have been taken but only 3 Safety Relief Valves (SRV) can be opened and no other means of depressurization is available.

Which of the following describes the consequences of this failure?

a. Steam removal rate from the core is not adequate to ensure adequate decay heat removal exists.

b. The pressure reduction rate will not allow low pressure injection systems to inject soon enough to recover level before core uncovery occurs.

c. Steam removal rate during a LOCA is not adequate to prevent exceeding the drywell design pressure.

d. The pressure reduction rate will not allow low pressure injection systems to inject prior to reaching the Minimum Steam Cooling RPV Water Level.

43 Given the following conditions:

- Unit 3 is in Mode 4 for a scheduled 6 day maintenance outage on drywell cooling
- A loss of shutdown cooling has occurred
- Shutdown cooling will not be restored for several hours
- Reactor water level +23 inches and steady
- Reactor water temperature 195 degrees F and rising

Select the ON-125, "Loss of Shutdown Cooling", action REQUIRED for these conditions.

a. Establish a vent path to the drywell via the reactor head vents.

b. Raise reactor water level to 30 inches to promote natural circulation.

c. Enter and take actions in accordance with T-102, "Primary Containment Control".

d. Verify the primary containment is in an Operable condition in accordance with GP-2.

44 Given the following conditions:

- Unit 2 is in Mode 5
- The Mode Selector Switch is in "Refuel"
- The Refueling Platform is over the spent fuel pool
- A fuel bundle has been loaded on the Main Hoist and raised out of the fuel pool storage rack
- The Unit Reactor Operator has just received a control rod block

What additional action was taken to cause this rod block?

a. The Refueling Platform operator raised the Main Hoist to the "full up" position.

b. The Unit Reactor Operator placed the Mode Selector Switch in "Startup/Hot Standby".

c. The Refueling Platform operator moved the platform over the reactor vessel.

d. The Unit Reactor Operator has selected, but NOT withdrawn, a single control rod.

45 Given the following conditions:

- Unit 3 is operating at 50% power
- Instrument nitrogen has been lost
- The instrument air backup valves to instrument nitrogen did NOT open
- DC power has been lost to the MSIV pilot solenoids
- No operator actions have been taken

What will be the expected Main Steam Isolation Valve (MSIV) response? (NOTE: Consider ONLY the response of the MSIVs for this question.)

a. The Inboard MSIVs will eventually drift closed, the Outboard MSIVs will remain open.

b. All eight MSIVs will remain open.

c. The Outboard MSIVs will eventually drift closed, the Inboard MSIVs will remain open.

d. All 8 MSIVs will close.

46 Unit 2 is operating at 100% power with all systems operating as designed. A failure in the Maximum Combined Flow Limit potentiometer causes its setpoint to lower to 80%.

Which of the following is the expected INITIAL response of the Electro-Hydraulic Control System for this failure?

a. The Main Turbine will trip from a power/load imbalance.

b. The Turbine Control Valves will close to a steam flow of 80%. The Turbine Bypass Valves will open to a steam flow of 20%.

c. The Load Set potentiometer will runback to maintain turbine load matched with steam flow.

d. The Turbine Control Valves will close to a steam flow of 80%. The Turbine Bypass Valves will remain closed.

47 Given the following conditions:

- Unit 2 is operating at 100%
- All plant systems have been operating as designed
- Reactor Feedwater Pump (RFP) speeds are all approximately 4150 rpm
- A transient occurs causing a single Condensate Pump trip
- Concurrent with this trip, a loss of the "B" RFP control signal occurs

Which of the following describes the final status of the RFPs for these conditions?

a. "A" and "C" RFP speed will remain constant and the "B" RFP maximum speed will be set by a limiter.

b. "A", "B" and "C" RFP maximum speeds will be set by a limiter.

c. "A" and "C" RFP maximum speed will be set by a limiter and the "B" RFP speed will remain constant.

d. "A", "B" and "C" RFP will all receive a continuous lower speed signal.

48 With Unit 2 at 100% power a loss of signal from all three Narrow Range level indicators occurs.

What is the expected plant response for these conditions?

a. Actual water level will be controlled approximately 6 inches below normal.

b. Actual water level will be controlled at approximately 33 inches.

c. The Master Level Control station will default to a -165 inch level setpoint.

d. The Master Level Control station will default to a 23 inch level setpoint.

49 Which of the following describes how the Reactor Building Ventilation/Standby Gas Treatment (SBGT) Systems operate with SBGT in service under non-emergency conditions (no Group III isolation signal present)?

With no Group III isolation signal present:

a. the SBGT Fan Vortex Dampers are throttled as needed from the control room.

b. the Reactor Building and Refuel Floor to SBGT Flow Dampers (PO-20477-1 & 2) are full open.

c. the SBGT Fan Vortex damper positions to automatically control pressure.

d. the Reactor Building and Refuel Floor to SBGT Flow Dampers (PO-20477-1 & 2) are throttled locally to control pressure.

50 Given the following conditions:

- Unit 2 is operating at 75% power
- The "A" Core Spray Pump is running for a surveillance
- The E212 Breaker has just opened
- All plant systems respond as designed

The "A" Core Spray Pump:

a. continue to run if bus voltage remains greater than 30% during the fast transfer.

b. will trip on undervoltage but will automatically restart if the Core Spray system receives an initiation signal.

c. continue to run if the fast transfer is completed in less than 0.25 seconds.

d. will trip on undervoltage and the breaker control switch must be placed to "Trip" and then to "Close" to restart the pump.

51 Given the following conditions:

- Both Units are operating at 100% power
- All systems are operating as designed in their normal lineups
- While applying a clearance the Plant Reactor Operator (PRO) opens the E-212 breaker

Which of the following describes the actions that must occur to reenergize the E-12 4KV Bus?

a. The PRO will have to close the E-312 breaker.

b. The E-1 Diesel Generator will automatically start and the E-12 Output Breaker will close.

c. The E-312 breaker will close provided the E-212 breaker switch remains in the "Normal After Close" position.

d. The E-1 Diesel Generator will automatically start and the PRO will have to close the E-12 Output Breaker.

52 Given the following surveillance results on the 2A 125 VDC Safeguard Battery with Unit 2 at 75% power:

- Maximum pilot cell float voltage - 2.19 volts
- Minimum pilot cell float voltage - 2.12 volts
- Maximum battery cell float voltage - 2.21 volts
- Minimum battery cell float voltage - 2.06 volts

What is the maximum permissible time Unit 2 may continue to operate in Mode 1? (Refer to attached Tech Specs.)

a. 2 hours

b. 12 hours

c. 14 hours

d. 31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—Operating

LCO 3.8.4 The following DC electrical power subsystems shall be OPERABLE:

a. Unit 2 Division I and Division II DC electrical power subsystems; and

b. Unit 3 Division I and Division II DC electrical power subsystems.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Unit 3 DC electrical power subsystem inoperable due to performance of SR 3.8.4.7 or SR 3.8.4.8.	-------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition A results in de-energization of a Unit 2 4 kV emergency bus or de-energization of a Unit 3 DC bus. ----------------------------	
	A.1 Restore Unit 3 DC electrical power subsystem to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Unit 3 DC electrical power subsystem inoperable for reasons other than Condition A.	--------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems—Operating," when Condition B results in de-energization of a Unit 2 4 kV emergency bus. -----------------------------	
	B.1 Restore Unit 3 DC electrical power subsystem to OPERABLE status.	12 hours
C. One Unit 2 DC electrical power subsystem inoperable.	C.1 Restore Unit 2 DC electrical power subsystem to OPERABLE status.	2 hours
D. Required Action and Associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	12 hours
	AND	
	D.2 Be in MODE 4.	36 hours
E. Two or more inoperable DC electrical power subsystems.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
SR 3.8.4.1 through SR 3.8.4.8 are applicable only to the Unit 2 DC electrical power subsystems. SR 3.8.4.9 is applicable only to the Unit 3 DC electrical power subsystems.

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is $\geq$ 123.5 V on float charge.	-----NOTE------ The 7 day Frequency is not applicable if the battery is on equalize charge or has been on equalize charge at any time during the previous 1 day. -------------- 7 days AND 14 days
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is $\leq$ 40 E-6 ohms.	92 days
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could potentially degrade battery performance.	12 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.4	Remove visible corrosion and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	12 months
SR 3.8.4.5	Verify battery connection resistance is ≤ 40 E-6 ohms.	12 months
SR 3.8.4.6	Verify each required battery charger supplies ≥ 200 amps at ≥ 125 V for ≥ 4 hours.	24 months
SR 3.8.4.7	-------------------NOTES------------------- 1. SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 once per 60 months when SR 3.8.4.8 envelops the duty cycle of the battery. 2. This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. ---	
	Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1, 2, or 3. However, credit may be taken for unplanned events that satisfy this SR. -- Verify battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months <u>AND</u> 12 months when battery shows degradation or has reached 85% of expected life with capacity < 100% of manufacturer's rating <u>AND</u> 24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.9	-------------------NOTE------------------- When Unit 3 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. --- For required Unit 3 DC electrical power subsystems, the SRs of Unit 3 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

LCO 3.8.5 The following DC electrical power subsystems shall be OPERABLE:

a. Unit 2 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown"; and

b. Unit 3 DC electrical power subsystems needed to support the DC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems—Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1	Declare affected required feature(s) inoperable.	Immediately
	OR		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.4.7 and SR 3.8.4.8. -- For required Unit 2 DC electrical power subsystems, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs
SR 3.8.5.2	-------------------NOTE------------------- When Unit 3 is in MODE 4 or 5, or moving irradiated fuel assemblies in the secondary containment, the Note to Unit 3 SR 3.8.5.1 is applicable. -- For required Unit 3 DC electrical power subsystems, the SRs of Unit 3 Specification 3.8.4 are applicable.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the station batteries shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1	Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>		
	A.2	Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>		
	A.3	Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells not within limits. <u>OR</u> One or more batteries with one or more battery cell parameters not within Category C limits.	B.1	Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	------NOTE----- The 7 day Frequency is not applicable if the battery is on equalize charge or has been on equalize charge at any time during the previous 4 days. --------------- 7 days AND 14 days
SR 3.8.6.2	Verify each battery cell meets Table 3.8.6-1 Category B limits.	92 days AND Once within 24 hours after battery discharge < 100 V AND Once within 24 hours after battery overcharge > 145 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 40°F.	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameter Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMIT FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ½ inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ ½ inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[b][c]	≥ 1.195	≥ 1.195 <u>AND</u> Average of all connected cells > 1.205	Not more than 0.020 below average of all connected cells <u>AND</u> Average of all connected cells ≥ 1.190

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when on float charge and battery charging current is < 1 amp.

(c) A battery charging current of < 1 amp when on float charge is acceptable for meeting specific gravity limits following a battery recharge for: 1) a maximum of 30 days if a deep discharge did not occur; and 2) a maximum of 180 days if a deep discharge did occur. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to the expiration of the applicable allowance.

53 Given the following conditions:

- The E-42 4KV Bus has lost power
- The fast transfer and Diesel Generator start both failed to occur automatically
- The E-4 Diesel Generator (DG) was started with the "Quick Start" pushbutton
- The E-42 breaker is closed and the DG is now carrying all the loads on the E-42 4KV Bus

Which of the following describes the current Mode of operation of the DG and what is required synchronize the DG back to the Grid?

The E-4 DG is operating in:

a. Parallel, the DG Quick Start pushbutton must be pressed again and synch must be completed within 3 minutes.

b. Unit, the DG Quick Start pushbutton must be pressed again and synch must be completed within 3 minutes.

c. Parallel, the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes.

d. Unit, the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes.

54 Given the following conditions:

- Unit 2 is operating at 10% power
- The E-3 Diesel Generator (DG) Start Mode Selector Switch is in "Auto"
- The E-3 DG Control Switch is in "Pull-To-Lock" for a test
- The E-32 4KV Bus has lost power with a failure to Fast Transfer
- A loss of all drywell cooling has resulted in a 2.5 psig drywell pressure

What is the status of the E-3 DG for these conditions?

a. Running with the E-32 Output Breaker open

b. Tripped with normal post-trip alarms in

c. Running with the E-32 Output Breaker closed

d. Ready for a start with the "Not In Auto" alarm in

55 During a lowering instrument air pressure transient ON-119, "Loss Of Instrument Air", directs a rapid plant shutdown if pressure is not stable above 75 psig.

How will a continued loss of instrument air impact the plant if it stays at power?

a. The chilled water to the Drywell Fans will be lost causing a drywell pressure rise.

b. The Reactor Building Ventilation system will no longer support normal building heat loads.

c. The Inboard Main Steam Isolation Valves will fast close (3 to 5 seconds).

d. The operability of plant emergency core cooling injection systems will be affected.

56 Which of the following is the Technical Specification limit value that is changed during single loop operation?

a. The Average Planar Linear Heat Generation Rate thermal limit

b. The size of the Immediate Exit region of the Power/Flow Map

c. Minimum Critical Power Ratio Safety Limit

d. The Temperature/Pressure limits for heatups and cooldowns

57 Which of the following describes how to place the Main Control Room Emergency Ventilation System (MCREV) in service from the Control Room?

a. Both MCREV System initiate pushbuttons must be pressed simultaneously.

b. Two of the four Control Room fresh air supply duct radiation monitors must be placed in "Test".

c. The fans and dampers must be manipulated via the Control Room controls.

d. The running Control Room Fresh Air Supply fan must be stopped.

58 Given the following conditions:

- Unit 2 was operating at 75% power when the "A" Recirculation Pump tripped
- Reactor power is 48%
- Calculated Total Core flow is 36%
- The Immediate Operator Actions of OT-112, "Recirculation Pump Trip" have been completed

Select the required operator actions for these conditions. (Power/Flow Map is attached)

a. Raise "B" Recirculation Pump speed to reach 45% core flow.

b. Lower the "B" Recirculation Pump speed to lower thermal power to less than 34%.

c. Restart the "A" Recirculation Pump to reach 50% core flow.

d. Scram and enter T-100, "Scram".

59 Unit 2 plant parameters have stabilized following a trip of the "A" Recirculation Pump from 100% power. The "B" Recirculation Pump is running at 1150 rpm and indicated core flow is 40%.

Which of the following describes the validity of total indicated core flow for these conditions?

Total indicated core flow:

a. is accurate if the "B" Recirculation Pump speed is greater than 40%.

b. must be raised by two times the "A" Recirculation Loop indicated flow.

c. is accurate if the "A" Recirculation Loop indicated flow is less than 3.0 Mlbs/hr.

d. must be reduced by two times the "A" Recirculation Loop indicated flow.

PBAPS POWER FLOW OPERATION MAP

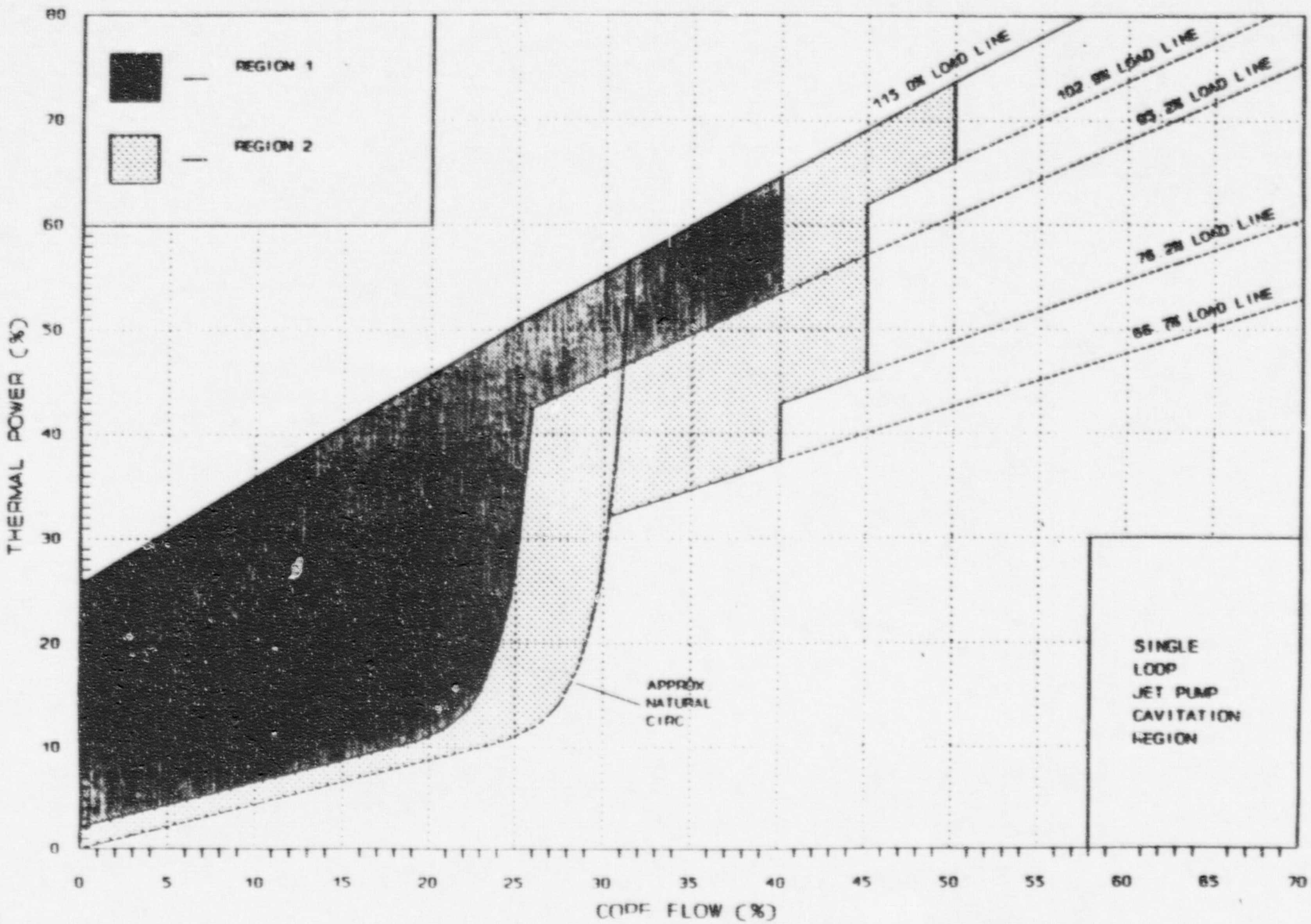


FIGURE 1

60 Given the following conditions:

- Unit 2 is at 2% power with the Mode Selector Switch in "Startup/Hot Standby"
- Main turbine shell warming is in progress
- The "C" Reactor Feedwater Pump is feeding the reactor
- While transferring from the Mechanical Vacuum Pump to the Steam Jet Air Ejectors, main condenser vacuum begins to lower
- No operator actions are taken

What will be the cause of the reactor scram?

a. Turbine Control Valve fast closure

b. Lowering Main Condenser vacuum (direct RPS trip)

c. Low reactor water level

d. High reactor pressure

61 Given the following conditions:

- Unit 2 was operating at 100% power
- Annunciator 220 F-5, "Inverter Trouble", was received indicating a loss of 20Y050
- The reactor was later scrammed and the turbine tripped

Which of the following is the reason why this failure requires reactor pressure control via the Safety Relief Valves?

The static inverter loss will:

a. cause a full Group I Main Steam Isolation Valve closure.

b. cause a "full open" signal to the Turbine Bypass Valves requiring the EHC Pumps to be tripped to prevent a rapid depressurization.

c. result in a closure of the Inboard Main Steam Isolation Valves.

d. result in a loss of Turbine Bypass Valve opening capability.

62. Given the following conditions:

- A loss of off-site power has occurred
- The E-1 and E-4 Diesel Generators (DG) are running
- The E-43 4KV bus has an overcurrent lockout
- No DG cooling water is available
- Drywell pressure is 3.6 psig and slowly rising

Why are jumper installations in the Control Room the PREFERRED method for shutting down the two diesel generators?

a. This bypasses the 10 minute timer on the MCA signal enabling the DG Control Switch "Pull-To-Lock" position.

b. Local methods of DG shutdown are disabled for these conditions.

c. The DG shutdown actions need to be completed as quickly as possible.

d. Use of the DG Control Switch "Pull-To-Lock" position will not allow a restart should cooling be restored.

63. Given the following conditions on Unit 2:

- With power at 22%, a loss of Stator Cooling occurred
- All automatic actions occurred as designed
- The turbine did not trip
- The Immediate Operator Actions of OT-113, "Loss Of Stator Cooling", have been completed
- There is no time estimate for restoration of Stator Cooling

The decision on if and when to trip the Main Turbine is based upon:

a. stator cooling water conductivity at the start of the transient.

b. the rate of increase of stator temperatures after the runback is complete.

c. the current plant location on the power to flow map.

d. final main generator field (amps) after the runback has gone to completion.

64 Given the following conditions on Unit 2:

- A scram has occurred due to a loss of feed
- Reactor water level reached -35 inches and is recovering using HPCI
- Nine control rods did not insert and remain at Notch "48"
- The Control Room Supervisor has entered T-100, "Scram" and T-99, "Post Scram Restoration"

Which of the following describes when the Unit Reactor Operator shall initiate ARI?

ARI shall be initiated:

a. as part of the expected actions directed by OM-P-16.1:5, "OSPS Reactor Operator Response To Reactor Scram".

b. as part of the "verify the scram" actions of T-100, "Scram".

c. when directed by T-99, "Post Scram Restoration".

d. when directed by GP-4, "Manual Reactor Scram".

65 Following a reactor scram, the Unit Reactor Operator reported that all APRMs are downscale. Later, the Control Room Supervisor (CRS) directed all control rods be verified to be inserted to or beyond Notch "02".

Which of the following describes why the CRS needs this information?

The CRS:

a. will direct boron injection (Standby Liquid Control) if this is not true.

b. is assured the reactor is shutdown and will remain shutdown during the ensuing cooldown.

c. will exit T-101, "RPV Control" and enter T-117, "Level/Power Control", if this is not true.

d. is assured the Heat Capacity Temperature Limit will not be exceeded.

66 Given the following conditions on Unit 2:

- A Main Steam Isolation Valve closure from 100% power has occurred
- Reactor pressure is 1050 psig
- Instrument nitrogen is not available to the Safety Relief Valves (SRVs)
- The pressure control leg of T-101, "RPV Control" requires a cooldown rate of less than 100 degrees F/hour and directs depressurization by "prolonged" SRV openings

Which of the following describes how the SRVs should be operated for these conditions?

The SRVs should be manually opened:

a. until 350 psig reactor pressure and then returned to the "Close" or "Automatic" position.

b. and closed to obtain pressure reductions in increments of slightly less than 100 psig until the reactor is depressurized.

c. and maintained open to obtain pressure reductions slightly less than the equivalent of 100 degrees F before shutting.

d. and maintained open until the reactor is depressurized or until pneumatic pressure is no longer available.

67 During a high reactor water level condition, the operator is directed to utilize LI-2(3)-2-3-86 to determine if main steam line flooding is occurring.

Which of the following must done to ensure this level indicator is providing an accurate reactor water level?

a. The indicated LI-86 level must be confirmed by an independent level indicator or computer point.

b. Reactor pressure must be compared to indicated level.

c. The drywell temperature and reactor pressure must be confirmed to be above the RPV Saturation Curve.

d. An adjustment must be made to account for recirculation pump flows.

68 Following a loss of shutdown cooling with no means of forced circulation available, ON-125, "Loss Of Shutdown Cooling" directs water level to be raised to greater than 50 inches.

This level is specifically meant to:

a. provide a larger margin to the low reactor water level shutdown cooling isolation setpoint.

b. establish a longer "time to boil" while lining up the alternate decay heat removal systems.

c. provide a larger margin of net positive suction head for restarting the RHR Pumps.

d. establish more favorable conditions to prevent stagnation of coolant in the reactor vessel.

69 Unit 2 is operating at 100% when the DCC-X Digital Feed Control System computer loses all of its Narrow and Wide Range level inputs.

What will be the expected response to these failures?

a. Control will be transferred to the DCC-Y computer and a "Feedwater Computer (X) Trouble" alarm will be received.

b. DCC-X will revert to a level default value of +23 inches and maintain reactor water level.

c. All three Reactor Feedwater Pumps will lockup and a "Digital Feedwater Field Instrument Trouble" alarm will be received.

d. Reactor water level will rise rapidly to the Reactor Feedwater Pump and Main Turbine trip setpoints.

70 Unit 2 was operating at 100% power when it experienced a steam leak in the drywell equivalent to 60 gpm. The Unit Reactor Operator has maximized drywell cooling in accordance with OT-101, "High Drywell Pressure".

Which of the following describes the bases for maximizing drywell cooling?

Maximizing drywell cooling:

a. reduces the individual loads on all of the drywell fans.

b. is done to prevent receiving high drywell pressure isolation and scram signals.

c. is designed to make up for the heat addition from a failure of one Recirculation Pump seal.

d. provides additional time for steam leak location and isolation as the source of the pressure rise.

71 Given the following conditions on Unit 2:

- A leak in the drywell has resulted in rising temperature and pressure
- All plant systems responded as designed as pressure exceeded 2.0 psig
- T-102, "Primary Containment Control", was entered for high drywell temperature
- T-223, "Drywell Cooler Fan Bypass" has been completed

With the drywell cooling fans returned to service, what is the source of cooling water, if any?

a. Reactor Building Closed Cooling Water

b. Drywell Chilled Water

c. Emergency service water

d. No cooling water is available, the fans run for circulation only

72 A loss of coolant accident is in progress on Unit 3. The scram was successful and current conditions are as follows:

- Reactor water level
  -- Narrow Range (LI-94A, B & C) indicates +5 to +8 inches
  -- Wide Range LI-85A & B) indicates -10 inches
  -- Shutdown Range (LI-86) indicates +10 inches
  -- Fuel Zone instruments indicate -25 inches
- Drywell temperature (TI-2-501)
  -- Point 126 indicates 270 degrees F
  -- Point 127 indicates 267 degrees F
- Reactor Building temperature (TR-2-13-139)
  -- Point 22 indicates 155 degrees F
- Reactor pressure is 200 psig

Which of the following instruments is NOT AVAILABLE for reactor water level indication?

a. Narrow Range

b. Wide Range

c. Shutdown Range

d. Fuel Zone

TABLE DW/T-1

RPV LEVEL INSTRUMENT STATUS

AN RPV LEVEL INSTRUMENT MAY BE USED TO DETERMINE RPV LEVEL ONLY WHEN THE FOLLOWING CONDITIONS ARE SATISFIED:

NOTES:

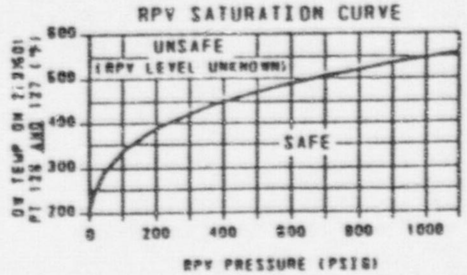
1. IF BOTH POINTS 126 AND 127 ARE AVAILABLE, THEN BOTH POINTS MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE

2. IF EITHER POINT 126 OR 127 IS NOT AVAILABLE, THEN THE REMAINING POINT MUST PLOT "SAFE" TO CONSIDER A LEVEL INSTRUMENT AVAILABLE

ALL RPV LEVEL INSTRUMENTS

THE COMBINATION OF DW TEMP AND RPV PRESS IS IN THE SAFE REGION OF THE RPV SATURATION CURVE

RPV SATURATION CURVE



AND

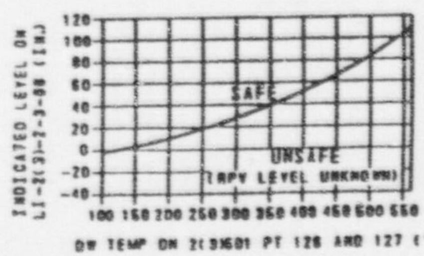
WIDE AND NARROW RANGE INSTS ONLY

FOR EACH OF THE INSTRUMENTS IN THE TABLE, THE INSTRUMENT READS ABOVE THE MIN INDICATED LEVEL OR THE TEMP NEAR THE DW REFERENCE LEG VERTICAL RUNS (TI-2(3)501 PT 126 AND 127) ARE BELOW THE MAX RUN TEMP.

INSTRUMENT	MIN INDICATED LEVEL IS ABOVE	OR	MAX RUN TEMP IS BELOW
NARROW RANGE	10 IN.	OR	450°F
WIDE RANGE	-120 IN.	OR	600°F

SHUTDOWN RANGE INST LI-2(3)-2-3-86 ONLY

LI-2(3)-2-3-86 READS ON THE SAFE SIDE OF THE CURVE



73 Given the following conditions:

- Unit 2 is operating at 75% power
- One Safety Relief Valve has just opened and will not close
- Torus water temperature is 95 degrees F and rising

What are the actions required by OT-114, "Stuck Open Safety Relief Valve", for these conditions?

a. Reduce power in accordance with "GP-9-2, "Fast Reactor Power Reduction", then when Recirculation Pumps are at minimum, place the Mode Selector Switch in "Shutdown".

b. Place the Mode Selector Switch in "Shutdown" prior to torus temperature reaching 120 degrees F.

c. Scram the unit in accordance with GP-4, "Manual Reactor Scram".

d. Reduce power in accordance with GP-9-2, "Fast Reactor Power Reduction", then place the Mode Selector Switch in "Shutdown" when torus temperature reaches 110 degrees F.

74 Following a reactor startup on Unit 3, the Reactor Engineer reports that the plant has experienced a reactivity anomaly.

What specific parameters must be monitored to determine the magnitude of this anomaly?

a. Core thermal power and nuclear instrumentation indicated power

b. Total control rod worth and individual control rod worth

c. Actual control rod density and core thermal power

d. Predicted control rod density and actual control rod density

75 With Unit 3 at rated power the Mode Selector Switch is placed in "Shutdown" to scram the reactor. The full core display "Blue" scram lights are not lit.

These conditions directly indicate that:

a. the backup scram valves did not de-energize to vent the scram air header.

b. the "Overtravel Beyond Full-In" reed switches have not picked up for each control rod.

c. a scram discharge volume hydraulic lock has occurred.

d. an electrical failure of the RPS Trip Systems has occurred.

76 Given the following conditions:

- Unit 2 has had a loss of all feedwater from 90% power
- 75 control rods did NOT insert on the scram
- At -195 inches, the Control Room Supervisor directed an Emergency Blowdown
- All injection to the reactor (except boron, CRD and RCIC) was terminated and prevented
- 5 Safety Relief Valves are open

Injection flow to the reactor will be reinitiated when:

a. reactor power is less than 3%.

b. reactor pressure is 230 psig

c. reactor water level is -210 inches

d. reactor pressure is 60 psig above torus pressure.

77 Unit 3 had been operating at full power for 412 days when a plant transient occurred requiring evacuation of the Main Control Room. Prior to leaving the Main Control Room all immediate actions directed by SE-1, "Plant Shutdown From The Remote Shutdown Panel", were completed.

What will be the expected approximate reactor pressure when the operators arrive at the remote shutdown panels? (Assume all systems are operating as designed.)

a. 940 psig

b. 1030 psig

c. 1135 psig

d. 1260 psig

78 Unit 2 is performing a plant cooldown in accordance with SE-10, "Alternative Shut Down". Reactor pressure is less than 75 psig. Cooldown is being accomplished using "Alternate Shutdown Cooling". How is the cooldown rate monitored while in this condition?

a. The local LPCI Pump discharge temperature.

b. The tailpipe temperature of the open Safety Relief Valve.

c. Reactor vessel skin temperatures at the main steam line level.

d. Main steam line saturated steam pressure converting to temperature per the SE-10 attachment.

79 A small steam leak with confirmed fuel failure has occurred in the Turbine Building. All T-104, "Radioactive Release Control", directed attempts to restart Turbine Building Ventilation have failed.

Which of the following is the impact of this failure?

a. The potential for a ground level release is greater.

b. The Turbine Building releases will be monitored but not treated.

c. The total off-site release will be greater.

d. The Turbine Building blowout panels will open.

80 With the Recirculation Pumps running at minimum speed, a loss of Reactor Building Closed Cooling Water occurs.

What are the restrictions on continued Recirculation Pump operation for these conditions?

In accordance with ON-113, "Loss of RBCCW", the Recirculation Pumps:

a. should be tripped immediately.

b. may continue to run as long as CRD purge is maintained on the seals.

c. should be tripped within one minute.

d. may continue to run as long as seal cavity temperatures remain within limits.

81 During a loss of instrument air the control rods begin to drift. Which of the following describes the direction of drift and what occurs to cause that movement?

The control rods drift:

a. in, similar to a normal insertion, because drive water flow and pressure both rise.

b. out, because a flowpath is opened to the top of the drive mechanism operating piston allowing reactor pressure to drift the rod out.

c. in, because the normal scram flowpath to and from the drive mechanism operating piston is opened allowing accumulator and reactor pressure to drift the rod in.

d. out, because a flowpath is opened from the bottom of the drive mechanism operating piston allowing accumulator pressure and gravity to drift the rod out.

82 Given the following conditions:

- Unit 2 is operating at 75% power
- All systems are operating as designed in automatic
- The Outboard MSIV in the "C" main steam line fails closed

Which of the following describes the expected plant response assuming no operator actions are taken?

a. Reactor pressure will spike, then stabilize at a value higher than its pre-transient value.

b. The remaining seven MSIVs close on high steam flow in the other steam lines.

c. Main generator output remains nearly constant and a MSIV closure half scram occurs.

d. Reactor pressure will spike, then return to its pre-transient value.

83 Given the following conditions:

- Unit 2 is in Mode 3
- Shutdown cooling is in service on the "A" RHR loop with a 8000 gpm flowrate
- Both Recirculation Pumps are shutdown
- Reactor water level by LI-86 (Shutdown Range) is out of service
- Reactor water level by LI-94 (Narrow Range) is 34 inches

In accordance with GP-12, "Core Cooling Procedure", the core is adequately cooled if: (Figure 1 of GP-12, "Core Cooling Procedure" is attached.)

a. present conditions are maintained.

b. reactor water level is raised to 40 inches.

c. "A" RHR flow is raised to 8,500 gpm.

d. reactor water level is raised to 37 inches.

84 Given the following conditions:

- Unit 2 is shutdown in Mode 4
- Shutdown cooling is in service on the "A" Residual Heat Removal (RHR) loop using the "A" RHR Pump
- Panel 20Y33 has just been lost

Which of the following describes the core heat removal capabilities for these conditions?

a. Alternate decay heat removal systems should be utilized.

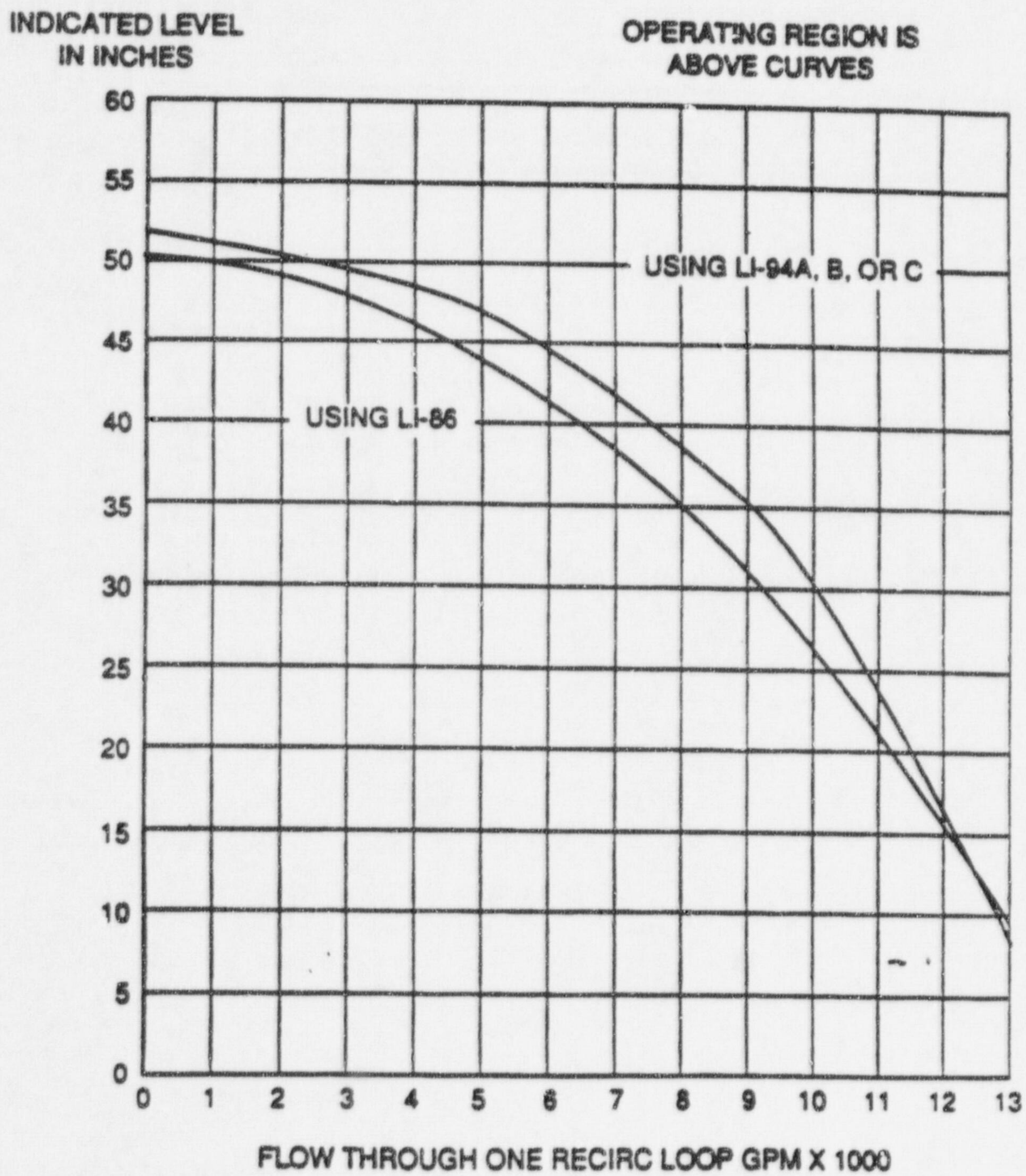
b. Shutdown cooling should be immediately transferred to the "B" RHR Loop.

c. The "A" RHR Pump should be reset and the "A" Loop returned to shutdown cooling

d. The RHR cross-tie valve should be opened to allow the "A" RHR Loop to return to the "B" Recirculation loop.

FIGURE 1

Vessel Level Vs. Flow Through One Recirc Loop



85 With Unit 2 operating at 90% power, annunciator 211 G-4, "CRD Charging Water Header High Pressure", is received.

Which of the following is the cause of this alarm and what is the expected impact on the plant?

a. The in-service CRD Flow Control Valve (AO-19) has failed closed resulting in rising Recirculation Pump seal temperatures.

b. The CRD Drive Water Pressure Control Valve (MO-20) has failed closed resulting in rising Recirculation Pump seal temperatures.

c. The in-service CRD Flow Control Valve (AO-19) has failed closed resulting in rising control rod drive mechanism temperatures.

d. The CRD Drive Water Pressure Control Valve (MO-20) has failed closed resulting in rising control rod drive cooling water flow.

86 Given the following conditions:

- Unit 2 is shutdown for refueling
- A Fuel Pool High Radiation alarm is received while moving a fuel assembly from the reactor vessel to its storage location
- The assembly was being lowered into a storage rack when the alarm occurred
- The assembly is NOT located near a radiation monitor

Which of the following describes what should be done with this fuel assembly prior to evacuation of the Fuel Floor?

The fuel assembly should immediately be:

a. returned to the reactor vessel and lowered back into its original location.

b. raised to above the top of the storage rack.

c. returned to the reactor vessel and left just above the upper grid.

d. lowered into its designated storage rack in the spent fuel pool

87 Given the following conditions:

- Unit 2 has experienced a loss of coolant accident with confirmed fuel failures
- Drywell and torus pressures reached 29 psig and sprays were initiated
- Sprays were NOT manually secured when pressure reached 2.0 psig
- Sprays did NOT automatically isolate at 1 psig

Which of the following is the expected impact on the plant for these conditions?

a. The drywell oxygen concentration may rise.

b. Torus water level indication will be unavailable.

c. The running Residual Heat Removal Pumps may cavitate.

d. Failure of the Reactor Building-Torus Vacuum Breakers will make the Reactor Building a High Radiation Area.

88 Which of the following MUST occur (and cannot be bypassed) to initiate Torus Sprays when required by T-102, "Primary Containment Control"?

a. A LPCI initiation signal must be present.

b. The Containment Spray Override 2/3 Core Coverage Switch (10A-S18) must be in "Override".

c. Reactor water level must be above -226 inches.

d. The Containment Spray Valve Switch (10A-S17) must be momentarily placed in "Manual".

89 With unit 2 operating at 100% power, one Safety Relief Valve (SRV) opened and did not reclose.

Which of the following tailpipe temperatures indicates the SRV is full open?

a. 212 degrees F

b. 250 degrees F

c. 315 degrees F

d. 544 degrees F

90 Which of the followin, ; is the reason why T-102, "Primary Containment Control", cautions against operating RCIC and HPCI with torus water temperatures above 190 degrees F?

a. The HPCI and RCIC Pump suctions will not transfer on low CST level.

b. The HPCI and RCIC turbine exhaust check valves will chatter.

c. The HPCI and RCIC Pumps will cavitate.

d. The torus will not condense all HPCI and RCIC turbine exhaust.

91 Given the following conditions:

- Unit 2 is operating at 100%
- One Safety Relief Valve has just opened and cannot be closed
- Drywell pressure and temperature are rising rapidly
- Drywell instrument run temperatures are rising rapidly
- Primary containment temperature control methods are unsuccessful

Assuming conditions continue to degrade, how would this impact the automatic initiation of the high pressure ECCS systems as reactor water level lowers?

High pressure ECCS systems:

a. initiate late because the wide range level indication will be reading 40 inches higher than actual level.

b. will not initiate because wide range level indication will be off-scale high.

c. initiate early because the wide range level indication will be reading 40 inches lower than actual level.

d. must be initiated by the operator because wide range level indication will be off-scale low.

92 Given the following conditions:

- Reactor pressure is 850 psig
- Torus water level is 17.5 feet
- Torus water temperature is 185 degrees F
- All controls rods are fully inserted

Which of the following would be expected if an emergency blowdown is performed for these conditions? See attached T-102 Curves.

a. Safety relief valves fail to operate.

b. Torus water will reach boiling sooner.

c. A reduction in Safety Relief Valve capacity (lbm/hour).

d. A direct discharge of steam to the containment.

93 Given the following conditions on Unit 3:

- A loss of coolant accident has occurred
- Torus water level is 11 feet
- Torus water temperature is 220 degrees F
- Torus pressure is 11.5 psig
- "A" RHR Loop flow is 9,500 gpm with the "A" Pump in operation
- "A" Core Spray loop flow is 3,500 gpm with the "A" Pump in operation
- The "B" RHR and "B" Core Spray Loops are not available
- Reactor water level has been recovered and is stable at +20 inches

Assuming reactor water level is maintained at it's current value, which of the following must be done? See attached T-102 ECCS Suction Requirement curves.

a. Reduce "A" Core Spray Pump flow to 2,500 gpm and raise "A" RHR Pump flow to 10,500 gpm.

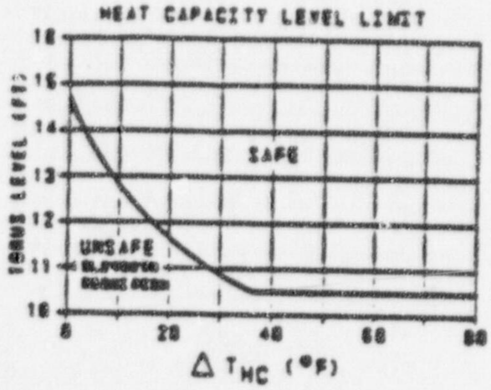
b. Start the "C" Core Spray Pump, throttle the "A" Core Spray Loop flow to 5000 gpm and throttle "A" RHR Loop flow to 8,000 gpm.

c. Secure the "A" Core Spray Pump and throttle the "A" RHR Pump flow to maintain 13,000 gpm.

d. Start the "C" Core Spray Pump, throttle the "A" Core Spray Loop flow to 7000 gpm and reduce "A" RHR Loop flow to 6000 gpm.

CURVE T/L-1

HEAT CAPACITY LEVEL LIMIT



ΔT_{HC} IN °F

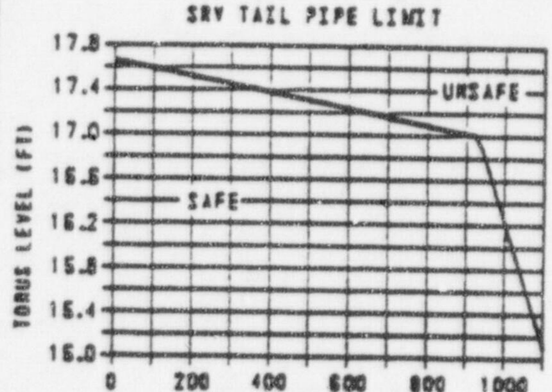
_____ °F HEAT CAPACITY TEMP LIMIT FROM CURVE T/T-1

MINUS (-) _____ °F TORUS TEMP

EQUALS (=) _____ °F ΔT_{HC}

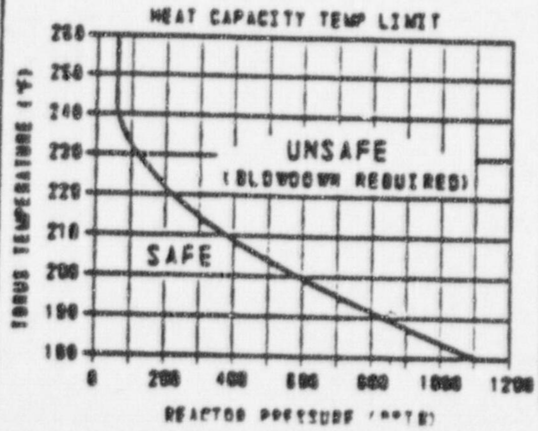
CURVE T/L-3

SRV TAIL PIPE LIMIT



CURVE T/T-1

HEAT CAPACITY TEMP LIMIT



REACTOR PRESSURE (PSIG)

94 High Pressure Coolant Injection (HPCI) is being operated from its Alternate Control Station in accordance with SE-10, "Alternative Shutdown". During a lowering torus water level transient, HPCI continues to run below the 9.5 foot torus water level limit imposed by T-102, "Primary Containment Control".

Which of the following is the consequence of continued HPCI operation?

a. The SRV Tail Pipe Limit will be exceeded.

b. HPCI will trip on high turbine exhaust pressure.

c. The Heat Capacity Temperature Limit will be exceeded.

d. The primary containment will exceed failure pressure.

95 Given the following conditions:

- Unit 2 has had a failure-to-scram (ATWS)
- Reactor water level cannot be determined
- 5 Safety Relief Valves have been opened
- The Condensate System is injecting
- Reactor pressure is 210 psig and lowering
- Reactor power is 2% and lowering

Which of the following describes the current plant status?

a. The steaming rate is less than the feed rate. The reactor is shutdown.

b. The current injection rate cannot maintain reactor pressure. Adequate core cooling does not exist.

c. Current reactor decay heat is insufficient to vaporize the injecting feedwater. Water level is at the top of active fuel.

d. Reactor water level is above the main steam lines. Adequate core cooling exists.

96 Unit 3 has experienced a loss of feed transient. Which of the following describes the means of core heat removal once reactor water level has lowered to -210 inches?

a. Opening the Safety Relief Valves reduces fuel cladding temperature via heat transfer to the steam passing the fuel assemblies.

b. Allowing reactor water level to continue to lower provides steam cooling until the Minimum Alternate Flooding Pressure is reached.

c. Opening the Safety Relief Valves causes the remaining water in the vessel to swell to above the top of active fuel to provide cooling.

d. Allowing reactor water level to continue to lower provides steam cooling until the Minimum Steam Cooling RPV Water Level is reached.

97 Given the following conditions:

- Unit 2 had been operating at power when a loss of feed occurred
- HPCI and RCIC initiated as designed and are restoring level
- A Reactor Building high temperature alarm is received
- T-103, "Secondary Containment Control", is entered and Reactor Building Ventilation is restored defeating isolations with T-222, "Secondary Containment Ventilation Bypass"

T-222 jumper installation is necessary to provide Reactor Building Ventilation to:

a. support Reactor Building cooling until reactor water level is above 1 inch.

b. give a more representative sample of Secondary Containment radiation levels.

c. support Reactor Building cooling with any Group III isolation signal present.

d. allow personnel access for investigation of the Reactor Building high temperature.

98 Given the following conditions:

- Unit 2 has entered T-103, "Secondary Containment Control" after receiving an alarm for high water level in the HPCI Room
- Condensate Storage Tank water level is dropping slowly
- No abnormal HPCI room equipment operation is apparent

In accordance with T-103, "Secondary Containment Control", which of the following is an acceptable means of determining that water level is at or above the "Action" level in the room?

a. Direct observation of the placard in the HPCI Room by opening the door from the RCIC Room.

b. Reactor Building Floor Drain Sump Hi-Hi Level alarm is received.

c. Direct observation of the water level in the stairwell outside the HPCI Room.

d. The HPCI Room water level computer point on SPDS is reading above 2 feet.

99 Given the following conditions:

- Unit 3 had a scram condition while at 100% power
- Several control rods failed to insert on the scram signal
- Reactor pressure is being controlled by the main turbine bypass valves
- No determination on the cause of the failure to scram has been made

Which of the following actions will result in motive force being applied to scram the rods WITHOUT REGARD for the cause of the failure?

a. De-energize the scram solenoids.

b. Isolate and vent the scram air header.

c. Vent the control rod drive overpiston area.

d. Reset the scram, drain the scram discharge volume and initiate a manual scram.

100. Given the following conditions on Unit 2:

- A failure-to-scram (ATWS) from 100% power has occurred
- One group of the Scram Pilot Solenoids, Channels "A" and "B", did not deenergize
- The Scram Discharge Volume is full
- T-101, "RPV Control", has been entered

Which of the following methods would be successful for control rod insertion with these conditions and the reason why it should be utilized?

a. T-213, "Deenergize Scram Solenoids", because it does not require resetting Reactor Protection System and Alternative Rod Insertion logic.

b. T-216, "Control Rod Insertion By Manual Scram Or Individual Scram Test Switches", because it does not require resetting Reactor Protection System and Alternative Rod Insertion logic.

c. T-213, "Deenergize Scram Solenoids", because it allows draining the Scram Discharge Volume prior to rod movement.

d. T-216, "Control Rod Insertion By Manual Scram Or Individual Scram Test Switches" because it allows draining the Scram Discharge Volume prior to rod movement.