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NRC Report

1

Question Data for Test: 2001 SRO

Question: 131

Unit 3 is in MODE 1 at 80% power.

- An applicable Tech Spec Surveillance with a 24 hour Frequency was last performed satisfactorily at 0900 on 1/1/01.
- The LCO Required Actions direct that the equipment be restored to OPERABLE status in 4 hours, or be in MODE 3 in 12 hours AND MODE 4 in 36 hours.

If a plant priority on Unit 2 prevents the surveillance from being performed, when is Unit 3 required to be in MODE 4?

- ☐ A. By 2100 on 1/3/01.
- ☐ B. By 0100 on 1/4/01.
- ☐ C. By 0300 on 1/4/01.
- ☑ D. By 0700 on 1/4/01

Explanation of Answer

A. Incorrect, 1/1/01 at 0900 + 24 hr frequency + 36 hours to MODE 4.
B. Incorrect. 1/1/01 at 0900 + 24 hr frequency + 6 hour grace + 36 hours to MODE 4.
C. Incorrect, 1/1/01 at 0900 + 24 hr frequency + 4 hrs restoration + 36 hrs to MODE 4.
D. Correct, 1/1/01 at 0900 + 24 hr frequency + 6 hour grace + 4 hour restoration + 36 hours to MODE 4.

Exam Level	Cognitive Level	Facility	Materials
SRO	Application	PBAPS	

KA Information

Tier: PWGs RO Grp: 1 SRO Grp: 1 RO Val: 2.9 SRO Val: 4.0 55.43 ☑

System:	Generic	Generic
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.12	Ability to apply technical specifications for a system.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam Question Source:

Ques Mod Met: Updated Dates and made minor wording enhancements

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Tech Specs	TS	3.0	.0-1,4	210	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Tech Specs	TS	1.3-1	1.3-3	210	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Introduction to Improved Tech Spe	PLOT-1800	III	16	7	2

2

Question Data for Test: 2001 SRO

Question: 132

You, as the Work Control Supervisor, have directed an Equipment Operator to perform a task which requires a key from the Shift Manager Key Cabinet.

In order for him to obtain the key, in accordance with the Nuclear Operations Manual, you must:

- [ ] A. Give the Equipment Operator verbal permission to get the key.
- [x] B. Issue the key personally with verbal permission from the Shift Manager.
- [ ] C. Give verbal permission for the Shift Operations Assistant (SOA) to issue the key.
- [ ] D. Have the Shift Operations Assistant (SOA) issue the key after obtaining verbal permission from the Shift Manager.

Explanation of Answer

B. Correct
A,C,D. Verbal permission is requried from the Shift Manager, and Shift Management is required to issue the key.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	N/A

KA Information

Tier: PWGs | RO Grp: 1 | SRO Grp: 1 | RO Val: 2.0 | SRO Val: 2.9 | 55.43 ☑

System:	Generic	Generic
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.13	Knowledge of facility requirements for controlling vital/controlled access.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
NOM (Chapters 0-5)	PLOT-1526	II.B.6.C	7	0	1.1

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual	NOM-C-5.3	3.0	2	0	

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Question Data for Test: 2001 SRO

Question: 134

A loss of both Standby Liquid Control Systems required Unit 2 to be placed in Mode 3. During the shutdown, the URO moved the reactor mode switch into the REFUEL position while attempting to place it into the SHUTDOWN position

The following conditions exists on Unit 2:

- Reactor mode switch position - REFUEL
- All rods full in.
- RPV pressure is 900 psig
- RPV level is +30 inches

Evaluate these plant conditions and determine Unit 2's current mode of operation.

- [x] A Mode 2
- [ ] B Mode 3
- [ ] C Mode 4
- [ ] D Mode 5

Explanation of Answer:

A. Correct-With head bolts tensioned and mode switch in REFUEL, Table 1.1-1 identifies Mode 2.
B. Incorrect-Mode switch must be in Shutdown Position.
C. Incorrect-Mode switch must be in Shutdown Position.
D. Incorrect-One or more RPV Head Closure Bolt must be less than fully tensioned.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	None

KA Information

Tier: PWGs   RO Grp: 1   SRO Grp: 1   RO Val: 2.8   SRO Val: 3.3   55.43 ☑

System:	Generic	Generic
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.22	Ability to determine mode of operation.

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Definitions	TS	Table 1.1-1	1.1-7	210	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Intro to ITS	PLOT-1800	II.F.1.n	21	7	4

4

Question Data for Test: 2001 SRO

Question: 135

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".

In accordance with NOM-C-9.1, "Independent Verification", which of the following describes the required action(s)?

- [x] 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.

Explanation of Answer

A. Correct answer.
B. Independent Verifier not authorized to modify COL steps.
C. Independent Verifier not authorized to "N/A" COL steps.
D. COL is correct, valve position is the problem. A-3 is not required here.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 1 SRO Grp: 1 RO Val: 3.4 SRO Val: 3.3 55.43 ☐

System:	Generic	
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.29	Knowledge of how to conduct and verify valve lineups.

Question Source Information

Ques Source: 1998 PBAPS NRC Exam Question Source:

Ques Mod Met: Updated procedural references.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Independent Verification	NOM-C-9.1	6.3.50	8	0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Chapt	PLOT-1527	11.B.4.a	12	0	1j

Question Data for Test: 2001 SRO

Question: 136

There is a caution tag affixed to the 20C004A Panel which states:

"Orifice Bypass Valve (MO-2-12-053) should be closed if Reactor Pressure is greater than 200 psig".

This Reactor Water Cleanup (RWCU) caution is to prevent:

- ☑ A overpressurizing the downstream piping.
- ☐ B exceeding system flow limits.
- ☐ C resin from sloughing off the RWCU filter demin elements.
- ☐ D water hammer in downstream piping.

Explanation of Answer

Procedure SO 12.1.A-2 caution states:

Restricting orifice RO-2-12-106 maximum flow rate is 185 gpm which prevents overpressurizing downstream piping. IF reactor pressure is equal to OR greater than 200 psig, THEN MO-2-12-053, "RWCU Orifice Bypass", should NOT be opened. IF reactor pressure is less than 200 psig THEN MO-2-12-053 may be opened to achieve high flows.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: PWGs | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.4 | SRO Val: 3.8 | 55.43 ☑

System:	Generic	Generic
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.32	Ability to explain and apply system limits and precautions.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Water Cleanup System St	SO 12.1.A-2	Caution	9	28	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Water Cleanup System	PLOT5012	II.D.4	15	1	4.a

6

Question Data for Test: 2001 SRO

Question: 137

Unit 3 is experiencing a hydraulic Anticipated Transient Without Scram (ATWS). The following plant conditions exist:

- Control rods are being manually inserted using T-220 "Driving Control Rods During Failure to Scram".
- "B" Standby Liquid Control (SBLC) pump is injecting boron into the reactor vessel.
- Reactor Engineering has been directed to complete a calculation to determine the reactor's shutdown condition.

Which one of the following conditions describes when the ATWS will be considered terminated in accordance with T-101 "RPV Control".

- [ ] A — 1 Control Rod is at position 12, 10 Control Rods are at position 02, all other rods are at position 00, 28% of the SBLC tank has been injected.
- [x] B — 27 Control Rods are at position 04, all other rods are at position 00, 2% of the SBLC tank has been injected.
- [ ] C — 3 Control Rods are at position 06, all other Control Rods are at position 00, 45% of the SBLC tank has been injected.
- [ ] D — 22 Control Rods positions are unknown, the SBLC tank has been fully injected into the vessel.

Explanation of Answer

TRIP Note #24 clearly defines what terminates an ATWS condition as: All rods inserted to or beyond the Maximum Subcritical Banked Rod Withdrawal Position (MSBRWP). OR See note 24

Note that for Unit 3 the MSBRWP is "04". (Unit 2 is "02".) This evaluates a unit difference.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	N/A

KA Information

Tier: PWGs — RO Grp: 2 — SRO Grp: 2 — RO Val: 3.1 — SRO Val: 3.3 — 55.43 ☑

System: Generic

KA Group Num: 2.2 — Equipment Control

KA Detail Num: 2.2.3 — (Multi-Unit) knowledge of the design, procedural, and operational differences between units.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met	

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control	T-101	Note #24	1	17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TRIP procedures	PLOT-1560			8	11

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TRIP Curves, Tables and Limits		Appendix 1	1	4	

7

Question Data for Test: 2001 SRO

Question: 138

A Post Maintenance Test (PMT) requires the performance of a portion of a Surveillance Test (ST) to stroke time a valve following maintenance to prove OPERABILITY. The Control Room Supervisor notes that the acceptance criteria for valve stroke time needs to be changed due to a recent Inservice Testing (IST) Program revision.

Which of the following is the MINIMUM action required to use the ST to complete this PMT.

- [ ] A — A "Partial Procedure Use Change".
- [ ] B — A "Permanent Revision Temporary Change".
- [ ] C — A "Single Use Temporary Change".
- [x] D — A "Procedure Revision".

Explanation of Answer: Temporary changes, partial procedure changes may not be made for changes to ST acceptance criteria because they change the intent of the procedure, the ST would have to be revised.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	

KA Information

Tier: PWGs — RO Grp: 2 — SRO Grp: 2 — RO Val: 2.3 — SRO Val: 3.3 — 55.43 ☑

System: Generic

KA Group Num: 2.2 — Equipment Control

KA Detail Num: 2.2.6 — Knowledge of the process for making changes in procedures as described in the safety analysis report.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam — Question Source:

Ques Mod Met: Minor wording enhancements

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Temporary Procedure Change	A-3	Exhibit A-3-1	1	0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Administrative Procedures	PLOT1570	II.B.1.e.1	7	15	1

8

Question Data for Test: 2001 SRO

Question: 141

The HPCI System Manager wishes to perform a diagnostic activity on the HPCI High Steam Flow Isolation Logic. The activity will involve lifting leads, checking electrical continuity and potentially cleaning and tightening of electrical connections. The activity is expected to take 1 hour. HPCI is considered to be Tech Spec INOP due to being isolated with the Aux Oil pump in "Pull to Lock". A 50.59 safety review has determined there is no un-reviewed safety question.

Which of the following procedures is required to control this activity?

- [ ] A. A-C-023 "Plant Evolution/Special Test (PEST) Program"
- [x] B. A-C-041 "Troubleshooting, Rework and Testing (TRT) Control Process"
- [ ] C. A-C-025 "Fix it Now (FIN) Process"
- [ ] D. MOD-C-7 "Temporary Plant Alteration (TPA)"

Explanation of Answer

A. Incorrect, activity is NOT an infrequently performed complex test or evolution which may place the plant equipment and operation outside bound of normal procedures.
B. Correct, A-C-41 and AG-CG-41, 4.4.1 addresses all listed activities.
C. Incorrect, FIN process does not include troubleshooting.
D. Incorrect, this activity does not involve a temporary alteration to the plant.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	None

KA Information

Tier: PWGs | RO Grp: 2 | SRO Grp: 2 | RO Val: 2.2 | SRO Val: 3.3 | 55.43 ☑

System:	Generic	
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.20	Knowledge of the process for managing troubleshooting activities.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met: Minor wording enhancements.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Troubleshooting, Reqork and Testi	AG-CG-41	4.4.1	4	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Administrative Procedures	PLOT-1570	II.B.1.e.6	9	14	16

Question Data for Test: 2001 SRO

9

Question: 142

Following a maintenance activity on MO-2-23-057 "HPCI Torus Suction Outboard", a partial ST-O-023-301-2 "HPCI Pump, Valve, Flow and Unit Coolers Functional and Inservice Test" was performed.

The initial stroke time for the valve was in the Action Range. The valve was stroked three additional times and was then within the acceptable limits of the ST.

The status of MO-2-23-057, "HPCU Torus Suction Outboard" valve is:

- [x] A NOT operable and the valve should be examined to determine the root cause.
- [ ] B NOT operable and a complete ST-O-023-301 should be performed to determine HPCI operability.
- [ ] C operable since the initial stroke test of the valve is NOT required as long as the second stroke time is within acceptable limits.
- [ ] D operable since, for post maintenance testing, the valve need only stroke full open and full closed.

Explanation of Answer: NOM-C-11.1 Section 4.9 states repeated "coaxing" of the valve can not be used to show operability and that test failures should be examined to determine the root cause and correct the problem before resumption of testing.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	

KA Information

Tier: PWGs   RO Grp: 2   SRO Grp: 2   RO Val: 2.3   SRO Val: 3.5   55.43 ☑

System:	Generic	Generic
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.21	Knowledge of pre and post maintenance operability requirements.

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Opera	NOM-C-11.1	4.9	17	0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Chapt	PLOT1528	II.B	5	0	

10

Question Data for Test: 2001 SRO

Question: 143

Unit 2 has been operating at full power when a loss of feedwater heating event occurs. The URO reports that maximum value of MFLCPR from an OFFICIAL 3D P1 edit is 1.19. The current MCPR operating limit from the Core Operating Limits Report (COLR) is 1.30. The scram times are within Technical Specification limits and all other equipment is operating normally.

Use the attached Technical Specifications to determine the correct response for this value of MFLCPR.

- [ ] A — NO actions are required, MCPR is within limits.
- [ ] B — Investigate an error with the 3D P1 Program, CPR should not be affected by a loss of feedwater heating.
- [ ] C — Restore MCPR within limits within two hours OR reduce thermal power to <25% RTP within four hours.
- [x] D — Restore MCPR within the safety limit value within two hours AND insert all control rods within two hours.

Explanation of Answer

$$MFLCPR = \frac{MCPR\ LCO}{MCPR\ ACT}$$

Therefore, (MFLCPR)(MCPR ACT) = (MCPR LCO) therefore

$$MCPR\ ACT = \frac{MCPR\ LCO}{MFLCPR} = \frac{1.30}{1.19} = 1.09$$

Exam Level	Cognitive Level	Facility	Materials
SRO	Application	PBAPS	Tech Spec Sections 2.0 and 3.2.2.

KA Information

Tier: PWGs — RO Grp: 2 — SRO Grp: 2 — RO Val: 3.4 — SRO Val: 4.1 — 55.43 ✓

System:	Generic	Generic
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.22	Knowledge of limiting conditions for Operations and Safety Limits.

Question Source Information

Ques Source: New — Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Fuel Design, Thermal Limit Applica	PLOT1870	II.D.1.f,g,h	23	4	8

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Fuel Design, Thermal Limit Applica	PLOT1870	II.D.1.f,g,h	23	4	9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Fuel Design, Thermal Limit Applica	PLOT1870	II.D.1.f,g,h	23	4	11

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Tech Spec COLR		2.0			

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the Plant Manager and the Vice President — Peach Bottom Atomic Power Station.

(continued)

2.0 SLs

2.2 SL Violations (continued)

2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the Plant Manager, and the Vice President—Peach Bottom Atomic Power Station.

2.2.5 Operation of the unit shall not be resumed until authorized by the NRC.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2

11

Question Data for Test: 2001 SRO

Question: 145

To return the plant to a stable condition during a transient, Operations personnel need to enter a High Radiation Area that does not have an existing Radiation Work Permit (RWP).

Which of the following will meet the MINIMUM requirements for an Equipment Operator to enter the area.

- [ ] A Must be accompanied by an Advanced Rad Worker (ARW) qualified individual.
- [ ] B Entry into the area is not permitted without the Radiation Protection Manager (RPM) permission.
- [x] C Must be accompanied by a Level II Radiation Protection Technician qualified individual.
- [ ] D Entry into the area is not permitted until activation of the Emergency Plan.

Explanation of Answer: HP-C-310 in a effort to return the plant to a stable condition a Level II (ANSI 3.1) RP Technician may act in lieu of a formal RWP to assist workers

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 3 SRO Grp: 3 RO Val: 2.6 SRO Val: 3.0 55.43 ✓

System: Generic

KA Group Num: 2.3 Radiation Control

KA Detail Num: 2.3.1 Knowledge of 10CFR20 and related facility radiation control requirements.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Radiation Work Permits	PLOT-1760	II.C	6	14	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Radiation Work Permit Program	HP-C-310	7.12		3	

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12

Question Data for Test: 2001 SRO

Question: 146

Peach Bottom Unit 2 is at 15% power. A drywell entry has been made to locate a source of rising drywell pressure. A failed diaphragm on a pressure regulator has been found and needs to be replaced.

As the Control Room Supervisor (CRS), use the As Low As Reasonably Achievable (ALARA) guidelines to determine which of the following methods should be directed to complete the repair. Consider only the radiation exposure aspects of this repair.

- [ ] A — Two individuals install temporary shielding for 30 minutes in a 200 mr/hr area, take 60 min. to complete the repair in a 20 mr/hr area, and remove the shielding for 20 minutes in a 200 mr/hr area.
- [ ] B — Two individuals take 60 minutes to complete the repair with no temporary shielding in a 200 mr/hr area.
- [x] C — Two individuals install temporary shielding for 20 minutes in a 200 mr/hr area, take 60 minutes to complete the repair in a 40 mr/hr area, and remove the shielding for 15 minutes in a 200 mr/hr area.
- [ ] D — One individual installs temporary shielding for 60 minutes in a 200 mr/hr area. Two individuals take 60 minutes to complete the repair in a 40 mr/hr area. One individual removes shielding for 30 minutes in a 200 mr/hr area.

Explanation of Answer:

A. 374 mr exposure
B. 400 mr exposure
C. 314 mr exposure - Correct answer.
D. 380 mr exposure

Exam Level	Cognitive Level	Facility	Materials
SRO	Application	PBAPS	Calculator

KA Information

Tier: PWGs | RO Grp: 3 | SRO Grp: 3 | RO Val: 2.5 | SRO Val: 2.9 | 55.43 ✓

System:	Generic	Generic
KA Group Num:	2.3	Radiation Control
KA Detail Num:	2.3.2	Knowledge of facility ALARA Program.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ALARA Program LP	PLOT1770	1.B		10	1

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ALARA Program LP	PLOT1770	1.B		10	2

13

Question Data for Test: 2001 SRO

Question: 147

Given the following conditions:

- A male, fully qualified radiation worker at Peach Bottom has just returned from 2 weeks of outage support at Three Mile Island (TMI).
- Total Effective Dose Equivalent (TEDE) received at TMI was 150 mrem.
- After a fall at home, the worker had an ankle x-ray estimated at 10 mrem exposure to the ankle.
- This workers' current TEDE from Peach Bottom for 2001 is 75 mrem.

What is the MAXIMUM annual non-emergency Total Effective Dose Equivalent (TEDE) that can be received at Peach Bottom for the remainder of 2001 WITHOUT exceeding the Federal Exposure Limits.

- [ ] A. 4765 mrem
- [x] B. 4775 mrem
- [ ] C. 4850 mrem
- [ ] D. 4925 mrem

Explanation of Answer: Federal Exposure Limit is 5000 mrem. 5000 mrem - 150 mrem (TMI) - 75 mrem (PBAPS) = 4775 mrem. The ankle x-ray is not considered to be occupational exposure.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 3 SRO Grp: 3 RO Val: 2.5 SRO Val: 3.1 55.43 ✓

System: Generic

KA Group Num: 2.3 Radiation Control

KA Detail Num: 2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Dosimetry Program	HP-C-106	7.1.10	3	6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Occupational Dose Limits for Adult	10CFR20.1201				

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Radiation Exposure Limits	PLOT-1730			12	2

14

Question Data for Test: 2001 SRO

Question: 148	Peach Bottom Unit 2 was operating at full power when the "A" Recirc Pump tripped. The following conditions exist: Reactor Power 68% - Calculated Single Loop Core Flow is 45% - Initial APRM flux noise level 2% - Final APRM flux noise level 3% - APRM flux oscillation period 10 seconds Use the attached "Peach Bottom Power Flow Operation Map" to determine the required Immediate Operator Action, if any, in accordance with OT-112 "Unexpected/Unexplained Change in Core Flow".
☐ A	NO immediate actions are required.
☐ B	Raise flow Pump "B" Recirc Pump until Region 2 is exited.
☑ C	Insert ALL GP-9-2 Appendix 1 Table 1 rods.
☐ D	Scram the reactor.
Explanation of Answer	A. Incorrect - Immediate Operation Action is to drive Table 1 rods. B. Incorrect - Raising recirc flow is a follow-up action and would not be completed until all Table 1 rods are inserted. C. Correct D. Incorrect - Power to Flow conditions don't require a scram and a THI condition does not exit as defined in OT-112.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	Power/Flow Map

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 4.3 SRO Val: 4.6 55.43 ✓

System:	Generic	
KA Group Num:	2.4	Emergency Procedures/Plan
KA Detail Num:	2.4.1	Knowledge of EOP entry conditions and immediate action steps.

Question Source Information

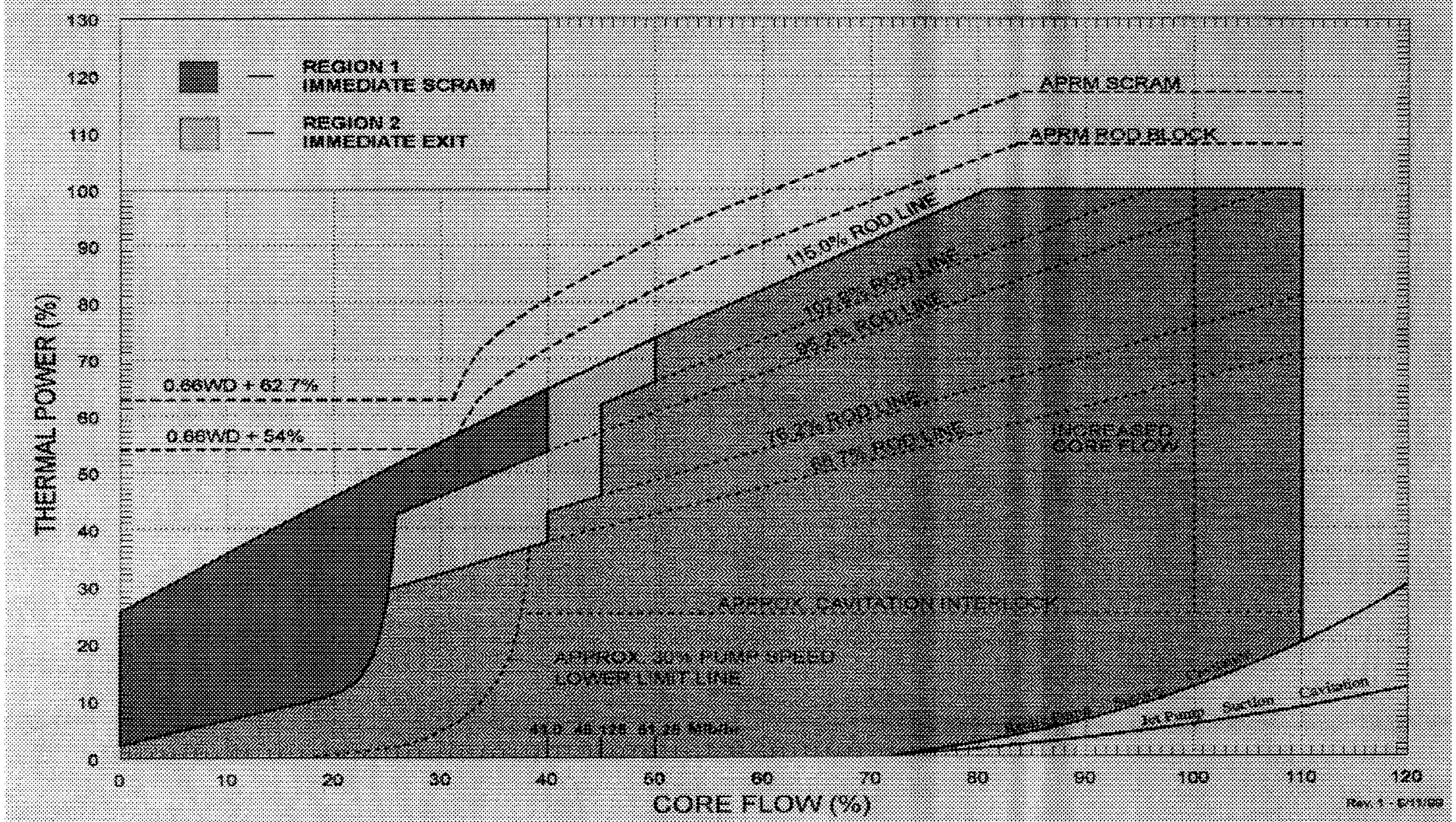
Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Unexpected/Unexplained Change i	OT-112			32	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Procedures	PLOT1540	II.B	6	6	3

PBAPS POWER FLOW OPERATION MAP



CM-1

15

Question Data for Test: 2001 SRO

Question: 149

Unit 3 was operating in MODE 1 at 50% power when a plant transient required the crew to scram the unit. The following conditions exist:

- All rods are inserted
- Reactor pressure dropped to 900 psig and has stabilized at approximately 940 psig
- Reactor level dropped to -20", then quickly recovered to its present value of +20" and going up
- HPCI auto started on a valid initiation signal and is injecting into the reactor vessel
- A Main Stack High Radiation Alarm is present
- A Turbine Building 165' elevation Area Radiation Monitor is alarming, reading 6 mr/hr

Select which of the following TRIP procedures should be entered and executed under these conditions.

- [ ] A Scram Condition (T-100)
- [x] B Primary Containment Control (T-102)
- [ ] C Secondary Containment Control (T-103)
- [ ] D Radioactive Release (T-104)

Explanation of Answer

A. Incorrect - T-100 should be exited, not executed, due to the T-101 entry condition on 2# DW pressure.
B. Correct - due to Drywell pressure as evidenced by the HPCI auto start.
C. Incorrect - ARM alarm is not in Secondary Containment
D. Incorrect - T-104 is not entered until a High High Main Stack Radiation alarm exists.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 4.0 SRO Val: 4.3 55.43 ✓

System:	Generic	
KA Group Num:	2.4	Emergency Procedures / Plan
KA Detail Num:	2.4.4	Ability to recognize abnormal indications for system operating parameter which are entry level condition.

Question Source Information

Ques Source:	1999 PBAPS NRC Exam	Question Source	
Ques Mod Met	Minor Enhancements		

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102		1	13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TRIP Procedures	PLOT-1560			8	1

16

Question Data for Test: 2001 SRO

Question: 150

A steam leak has occurred on Peach Bottom Unit 2. Drywell sprays have been properly placed in service using the "A" RHR Pump. The following conditions exist:

- Torus level 11.5 ft.
- Torus temperature 180 degrees F
- Drywell Bulk Average temperature 260 degrees F
- Drywell pressure is 15 psig and lowering
- Torus pressure is 14 psig and lowering
- "A" RHR Loop flow is 10,000 gpm

Reference the attached T-102 Curves (DW/T-2 and NPSH Curves) as required to select from the following the FIRST point at which Drywell Sprays would need to be terminated.

- [ ] A Immediately
- [ ] B When Drywell pressure lowers to 5 psig with Drywell bulk average temperature at 210 degrees F.
- [x] C When Torus pressure lowers to 3 psig.
- [ ] D When Torus pressure lowers to 2 psig.

Explanation of Answer: CAUTION #10 bases say that the operator should maintain NPSH unless directed to use a pump regardless of NPSH concerns. Drywell pressure must stay above 3 psig to maintain NPSH for this RHR flow and Torus temperature.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	T-102 Sheet 3 (NPSH Curves) DW/T-2 from T-102 (DW Spray Initiation Curve)

KA Information

Tier: PWGs | RO Grp: 4 | SRO Grp: 4 | RO Val: 3.3 | SRO Val: 3.4 | 55.43 ☐

System: Generic

KA Group Num: 2.4 — Emergency Procedures/EOPs

KA Detail Num: 2.4.20 — Knowledge of operational implications of EOP warnings, cautions, and notes.

Question Source Information

Ques Source: New

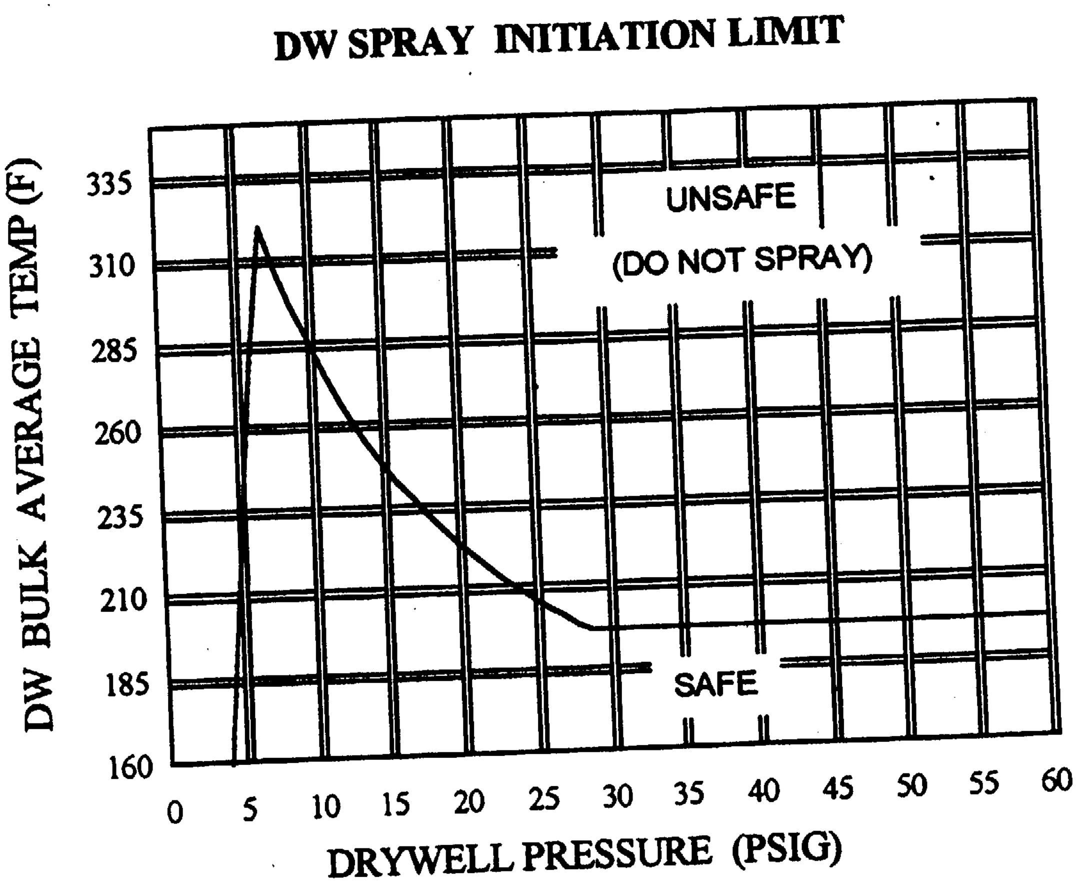
Question Source:

Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control (Bas	T-102 Bases			15	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TRIP Procedures	PLOT1560	II.C	10	15	11


DW SPRAY INITIATION LIMIT
DW BULK AVERAGE TEMP (F)
335
310
285
260
235
210
185
160
0
5
10
15
20
25
30
35
40
45
50
55
60
DRYWELL PRESSURE (PSIG)
UNSAFE
(DO NOT SPRAY)
SAFE

Question Data for Test: 2001 SRO

Question: 152

You are the Peach Bottom Emergency Director (ED) during an emergency on Unit 3. In accordance with ERP-200 "Emergency Director", select the following duty/responsibility that may be delegated?

Authorizing personnel to:

- ☐ A. issue and use Potassium Iodide (KI).
- ☐ B. exceed the Peach Bottom Administrative Dose Limits.
- ☐ C. exceed the Federal Emergency Radiation Exposure Limits.
- ☑ D. implement the Severe Accident Management (SAM) program.

Explanation of Answer:

A. Incorrect - Non-delegable per ERP-680.
B. Incorrect - Non-delegable per ERP-670.
C. Incorrect - Non-delegable per ERP-670.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 2.2 SRO Val: 4.0 55.43 ☑

System: Generic — Generic

KA Group Num: 2.4 — Emergency Procedures/Plan

KA Detail Num: 2.4.38 — Ability to take actions called for in the Facility Emergency Plan, including supporting or acting as Emergency Coordinator.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Director	ERP-200	3.0	2	16	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Radiation Exposure Gu	ERP-670	1.0	1	4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Control of Thyroid Blocking Potassi	ERP-680	1.0	1	6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Director Training	PEPP6010	II.C.1	4	3	6.a

18

Question Data for Test: 2001 SRO

Question: 153

Peach Bottom Unit 2 is operating at 100% power. The CRD Hydraulic System is aligned for normal operation with the "A" Flow Control Valve (AO-2-3-019A) in service.

The 70#-100# air line connection to the air operator for the "A" Flow Control Valve fails, causing the actuator to depressurize.

This loss of air will cause the "A" Flow Control Valve to fail:

- ☐ A — open, resulting in CRD Drive Water Header differential pressure dropping.
- ☐ B — open, resulting in CRD Drive Water Header differential pressure rising.
- ☑ C — closed, resulting in CRD Drive Water Header differential pressure dropping.
- ☐ D — closed, resulting in CRD Drive Water Header differential pressure rising.

Explanation of Answer: The flow control valves will fail closed on loss of air pressure to the actuator. The CRD Drive water differential pressure transmitter senses pressure downstream of the flow control valves. Drive water DP is directly related to the throttled position of the Drive Water Pressure Control Valve (MO-20), and flow from the FCV through MO-20. The loss of flow will cause DP to lower.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 2 | RO Val: 3.1 | SRO Val: 2.9 | 55.43 ☐

System: 201001 Control Rod Drive Hydraulic System

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: 201001A10 — CRD Drive Water Header Pressure

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
CRDH System Lesson Plan	PLOT-5003A	E.3.b	27	2	6.C

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Inst. Air	ON-119	Attachment 1	12	14	

19

Question Data for Test: 2001 SRO

Question: 155

You are the Startup SRO during a Unit 3 startup. Reactor pressure is 940 psig with 3 bypass valves open. The crew has entered ON-106 "Stuck Control Rod". In accordance with ON-106, you have directed the Reactor Operator to raise Control Rod Drive Pressure.

The procedure cautions you to minimize the time that CRD drive pressure is elevated to prevent:

- ☐ A — drifting control rods due to high drive water pressure differential.
- ☐ B — damage to CRDM collet fingers due to high pressure differential.
- ☐ C — drifting control rods due to high cooling water flow.
- ☑ D — damage to CRDM seals due to low cooling water flow.

Explanation of Answer

A. High drive water dP will not cause control rods to drift.
B. High drive water dP will not damage the collet fingers.
C. Cooling water flow will be low, not high, with elevated drive water pressure.
D. Correct answer. Seals overheat due to low flow.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	None

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 2 | RO Val: 3.4 | SRO Val: 3.8 | 55.43 ☑

System:	201001	Control Rod Drive Hydraulic System
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.32	Ability to explain and apply system limits and precautions.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Stuck Control Rod	ON-106	2.6 Caution	2	5	

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Question Data for Test: 2001 SRO

Question: 156

Peach Bottom Unit 2 is operating at a reduced power for a rod pattern adjustment with the following conditions:

- Core Flow 90%
- Reactor Power 95%
- Both recirculation pumps in service.

Failures cause the A and B Recirculation Pump speeds to rise resulting in a core flow increase to the APRM rod block setpoint.

Use the attached Power Flow Operation Map to determine the expected percent core flow with power at the rod block setpoint? Core flow rose to approximately:

- ☐ A 95%
- ☐ B 102%
- ☑ C 108%
- ☐ D 120%

Explanation of Answer

A. Incorrect-Intersection with 100% licensed power.
B. Incorrect-Misinterpreting Rod Line % as flow %.
C. Correct-Following 102.9% Rod Line to intersect with APRM rod block line.
D. Incorrect-Appropriate intersection with APRM Scram Line.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	Exhibit GP-5-1, PBAPS Power Flow Operation Map

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.5 SRO Val: 3.5 55.43 ☐

System: 202002 — Recirculation Flow Control System

KA Group Num: K3 — Knowledge of the effect that a loss or malfunction of the system will have on the following:

KA Detail Num: 01 — Core Flow

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

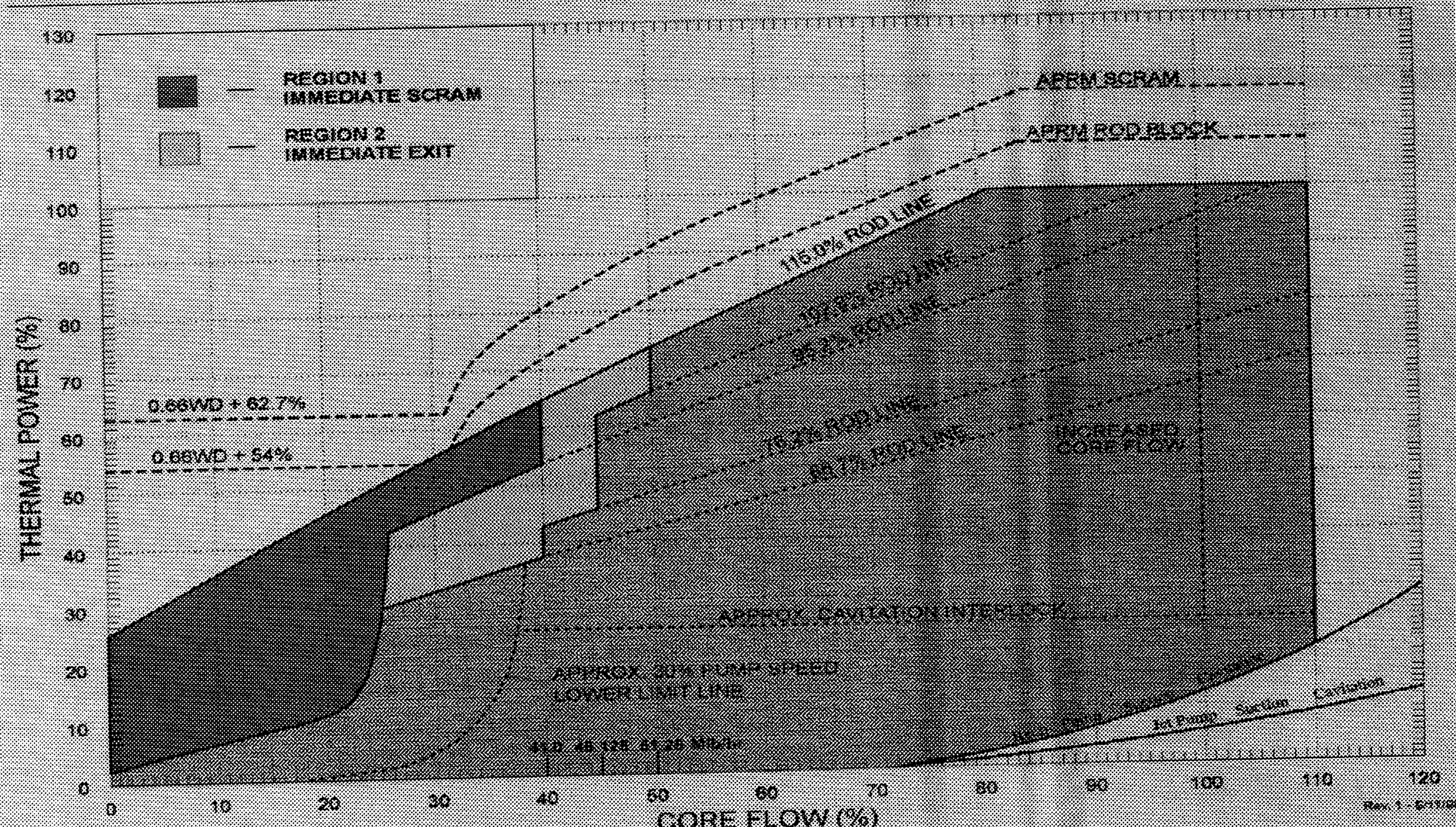
References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Power Flow Operation Ma	Exhibit GP-5-1		1	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT-5002	H	55	1	1.b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT-5002	H	55	1	3.a

PBAPS POWER FLOW OPERATION MAP



CM-1

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Question Data for Test: 2001 SRO

Question: 157

Peach Bottom Unit 2 was at 25% power while performing a shutdown when the following events occurred:

- House loads were manually transferred to off-site power
- The "A" Scoop Tube Positioner Trouble Alarm (214 E-2) and "A" Recirc Fluid Drive Scoop Tube Lock (213 A-3) alarmed due to high oil temperature.
- The Main Turbine tripped on a loss of Main Turbine lube oil pressure

Which of the following identifies the expected status of the "A" and "B" Recirc Pumps with no Operator action.

- ☐ A. BOTH pumps operating at 30% speed.
- ☐ B. "A" pump operating at pretransient speed, "B" pump operating at 30% speed.
- ☑ C. "A" pump tripped, "B" pump operating at the pretransient speed.
- ☐ D. BOTH pumps tripped

Explanation of Answer

A. Incorrect - The "A" pump trips on high MG oil temp
B. Incorrect - The "A" pump trips on high MG oil temp, the "B" does not runback to 30% speed.
C. Correct - "A" pump would receive a trip signal on High MG oil temp and "B" pump would continue to operate at its pretransient speed.
D. Incorrect - EOC-RPT trips are bypassed at this power level

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	NA

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.5 | SRO Val: 3.6 | 55.43 ☑

System:	202002	Recirculation Flow Control System
KA Group Num:	2.4	Emergency Procedures / Plan
KA Detail Num:	2.4.46	Ability to verify that alarms are consistent with the plant conditions.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 214 E-3	214 E-3		1	2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc / Recirc Flow Control	PLOT-5002	2.E.2	36	2	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 213 A-3	213 A-3		1	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc MG Set Scoop Tube Lockup	SO 2D.7.B-3	4	2	11	

22

Question Data for Test: 2001 SRO

Question: 159

A Dual Unit Design Bases Loss Of Coolant Accident (LOCA) event has occurred on Peach Bottom. The Unit 3 RHR "A" Loop Injection Valve (MO-3-10-25A) normal power suppy (E-134-R-C) has been lost.

Which one of the following describes the expected response of the Unit 3 RHR System with this power supply unavailable?

- ☐ A — BOTH loops align for injection with all four RHR pumps running.
- ☐ B — ONLY the "B" loop aligns for injection with all four RHR pumps running.
- ☑ C — BOTH loops align for injection with one RHR pump running in each loop.
- ☐ D — ONLY the "B" loop aligns for injection with one RHR pump running in each loop.

Explanation of Answer:
A. Incorrect-Only 2 pumps per unit start on dual unit LOCA.
B. Incorrect-Both loop MO25's will align and only 2 pumps start.
C. Correct-MO25 would swap to alternate power and open as designed.
D. Incorrect-Both loop MO25's will align.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 2.5 | SRO Val: 2.7 | 55.43 ☐

System: 203000 — RHR / LPCI: Injection Mode

KA Group Num: K2 — Knowledge of Electrical Power Supplies to the following

KA Detail Num: K2.02 — Valves

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
E134-R-C MCC or E134 Emer. L.C	AO 56E.2-3	4		7	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Logic	M-1-S-65		48/56	96	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT-5010	C	19	1	2.b

Question Data for Test: 2001 SRO

Question: 161

Peach Bottom Unit 3 has scrammed from 100% power and all Main Steam Isolation Valves are closed. RPV pressure is being maintained 950-1050 psig by manual operation of Safety Relief Valves. RPV level has been restored to +5" using HPCI and RCIC.

The Plant Reactor Operator (PRO) has been directed to place HPCI in the CST-To-CST mode to assist with pressure control. The PRO opens the "COND TANK RETURN" Valve (MO-3-23-24) fully and throttles open the "Full Flow Test" Valve (MO-3-23-21).

Select the answer that correctly describes the HPCI System response as these valves are opened.

HPCI pump:

- [ ] A discharge pressure lowers and speed rises.
- [ ] B discharge pressure and speed both rise.
- [x] C discharge pressure and speed both lower.
- [ ] D discharge pressure rises and speed lowers.

Explanation of Answer: As the test return valves are opened, system pressure decreases due to less flow resistance. System flow goes up. The flow control system closes the governor valve to maintain the flow controller setpoint. This results in speed decreasing.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.5 55.43

System:	206000	High Pressure Coolant Injection System
KA Group Num:	A3	Ability to monitor automatic operations of the system including
KA Detail Num:	A3.01	Turbine Speed: BWR-2, 3, 4

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Rapid Response Card	RRC 23.1-3	C	2	2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Manual Operation	SO 23.1.B-3	4.4	6	13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI System Lesson Plan	PLOT-5023	C.3	14	1	4.l

24

Question Data for Test: 2001 SRO

Question: 162

Peach Bottom Unit 2 has experienced a Loss of Off-Site Power (LOOP). The Emergency Diesel Generators have all started and are powering their 4KV busses. Due to a lowering reactor water level, the CRS directs you to use the "Arm and Depress" pushbutton to start the Core Spray system.

After arming and depressing "CS B INITIATION" pushbutton (14A-S10B), what is the expected response of the Core Spray system?

- ☐ A "A", "B" ,"C", and "D" Core Spray pumps start immediately.
- ☐ B "A", "B", "C", and "D" Core Spray pumps start after a time delay.
- ☑ C "B" and "D" Core Spray pumps start immediately.
- ☐ D "B" and "D" Core Spray pumps start after a time delay.

Explanation of Answer: For Core Spray "B" pushbutton starts ONLY "B" & "D" pumps. (On RHR, all 4 start.)
Time delays are not active when the 4 KV busses are powered from the Diesel Generators.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.8 SRO Val: 3.6 55.43 ☐

System:	209001	Low Pressure Core Spray System
KA Group Num:	A4	Ability to manually operate and/or monitor in the control room.
KA Detail Num:	A4.05	Manual Initiation Controls

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Spray	PLOT-5014	V.B.7	18	0	5.h

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Spray Logic	M-1-S-40		2, 3		

Question Data for Test: 2001 SRO

Question: 163

Peach Bottom Unit 2 is experiencing a Hydraulic Anticipated Transient Without Scram (ATWS) condition. The Standby Liquid Control (SBLC) System has been initiated and the "A" SBLC Pump is injecting into the vessel when a loss of all offsite power condition occurs. All four diesel generators start normally and load their buses.

One minute later, the Reactor Operator is directed to verify the status of the SBLC System. The operator should expect to see the "A" SBLC Pump:

- [x] A running with the squib valve continuity lights lit.
- [ ] B running with the squib valve continuity lights NOT lit.
- [ ] C NOT running with the squib valve continuity lights lit.
- [ ] D NOT running with the squib valve continuity light NOT lit.

Explanation of Answer: The pump should be running since its 480V Emergency Power Supply has been restored by the diesel generators. The squib valve continuity light will be lit even if the valves have fired as long as the pump is running.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 2.9 SRO Val: 3.1 55.43 ☐

System: 211000 Standby Liquid Control System

KA Group Num: K2 Knowledge of electrical power supplies to the following

KA Detail Num: K2.01 SBLC Pumps

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Liquid Control System	PLOT-5011	II.D.1.I	14	0	2.a

26

Question Data for Test: 2001 SRO

Question: 165

Peach Bottom Unit 2 is operating at 100% power when the 'A' Recirculation Pump Moore Controller Output instantaneously fails to maximum.

Select the expected plant response with no operator action.

- [ ] A — Reactor scram on low RPV level.
- [ ] B — 'A' Recirc Pump trips on high vibration.
- [x] C — Reactor scram on high flux.
- [ ] D — 'A' Recirc MG Set scoop tube locks

Explanation of Answer: Recirc speed rise causes void fraction in the core to drop with an immediate rise in reactor power to the scram setpoint.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 4.4 | SRO Val: 4.4 | 55.43

System: 212000 — Reactor Protection System

KA Group Num: A3 — Ability to monitor automatic operations of the system including

KA Detail Num: 212000A30 — Reactor Power

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Protection	PLOT-5060F	C.3.b.D	25	1	1a
Reactor Protection	PLOT5060F	C.3.b.D	25	1	3c

27

Question Data for Test: 2001 SRO

Question: 166

Peach Bottom Unit 2 is operating at full power with the Traversing In-Core Probe (TIP) System in service. A small steam leak occurs causing drywell pressure to rise to 3.5 psig. The TIP System has continued to operate normally.

Select the statement which describes the expected TIP System response, if any, to these conditions.

The TIP System:

- [ ] A — is responding correctly since it does not automatically isolate under these conditions.
- [x] B — should have isolated. A potential primary containment leakage path exists requiring operator action.
- [ ] C — should have isolated. Operator action is only required if the TIP Room (RB 135') conditions exceed the T-103, Secondary Containment Control Maximum Safe Levels.
- [ ] D — should have isolated. NO operator action is required since the drive cable provides an effective isolation boundary while inserted through the containment penetration.

Explanation of Answer:

A. Determines if candidate understands there is a GP II Isolation for TIP System.
B. Correct
C. Tip should isolate. Immediate Operator Action is required.
D. The cable does go through the containment penetration of concern but does not form a boundary or barrier to the leak path.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 3 | SRO Grp: 3 | RO Val: 3.4 | SRO Val: 3.7 | 55.43 ☐

System: 215001 — Traversing In-Core Probe

KA Group Num: A2 — Ability to (a) predict the impacts of the following on the system; & (b) based on predictions, use procedures to correct, control, or mitigate the consequences of abnormal conditions or mitigation...

KA Detail Num: A2.07 — Failure to Retract During Accident Conditions

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP System Isol Proc	SO 7F.7.A-2	4.1		2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP Lesson Plan	PLOT5007F	C		0	4a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP Lesson Plan	PLOT5007F	C		0	1e

28

Question Data for Test: 2001 SRO

Question: 168

During a reactor startup on Peach Bottom Unit 3, power is at 40% of rated. Upon selection of the next in sequence rod, an "A" Rod Block Monitor (RBM) fails to successfully complete its Null Sequence.

Which one of the following identifies the impact this failure will have on continued rod withdrawal?

Continued rod withdrawal will be:

- [x] A — prevented due to an INOP trip on the "A" RBM.
- [ ] B — prevented due to a Comparator trip on the "B" RBM.
- [ ] C — permitted due to an INOP trip on only one of the two RBMs.
- [ ] D — permitted due to only receiving a trouble alarm on "A" RBM.

Explanation of Answer

A. Correct - A failure to null gives you an INOP trip.
B. Incorrect - Flow Comp. Alarm is due to >10% difference between recirc loop flow indications.
C. Incorrect - Either RBM will generate a rod block.
D. Incorrect - Failure to null will generate an INOP trip.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 3.4 SRO Val: 3.5 55.43 ☐

System: 215002 Rod Block Monitor System

KA Group Num: K4 — Knowledge of this system design feature(s) and/or interlocks which provide for the following.

KA Detail Num: K4.01 — Prevent control rod withdrawal.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC (30C205R)	311 C-3		1	8	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PRNM/OPRM/RBM	PLOT-5060	D	36	1	4.a

Question Data for Test: 2001 SRO

Question: 170

A reactor startup is in progress on Peach Bottom Unit 2. Just after reaching the point of adding heat, a loss of Uninterruptable AC power (20Y50) occurs. Which one of the following identifies the ability to determine reactor power during this power loss?

Reactor power can:

- [ ] A be determined on the 20C005 panel ODAs and the 20C036 panel chassis.
- [ ] B be determined on the 20C005 panel ODAs ONLY.
- [x] C be determined on the 20C036 panel chassis ONLY.
- [ ] D NOT be determined on either the 20C005 panel ODAs or the 20C036 panel chassis.

Explanation of Answer

A. Incorrect - ODA's lose power.
B. Incorrect - ODA's lose power.
C. Correct - 20Y50 power C005 panel ODA's - 24 VDC powers C03G chassis.
D. Incorrect - Chassis powered from DC.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 2 RO Val: 3.6 SRO Val: 3.6 55.43 ☐

System: 215003 Wide Range Neutron Monitor (WRNM) System

KA Group Num: K3 Knowledge of the effect that a loss or malfunction of the system will have on the following.

KA Detail Num: K3.04 Reactor Power Indication

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Uninterruptible AC Power	ON-112-2	2	1	3	

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Question Data for Test: 2001 SRO

Question: 171

A reactor startup is in progress on Peach Bottom Unit 3. Power is on Range 2 of the WRNMs when a loss of power to the "A" WRNM chassis occurs.

Under these conditions, the failure will cause:

- [ ] A — a full reactor scram signal to be generated.
- [x] B — a rod block and a half scram to be generated.
- [ ] C — only a trouble alarm to be generated.
- [ ] D — the chassis to swap to its alternate power supply.

Explanation of Answer

A. Incorrect - Only one WRNM lost power.
B. Correct - WRNM logic to RPS is 1 out of 2 taken twice with 24 VDC lost all functional outputs from the drawer would occur INOP, short period etc. due to deenergize to trip design.
C. Incorrect - Also a rod block and RPS input.
D. Incorrect - WRNM does not have WLVPS like the APRMs.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS — RO Grp: 1 — SRO Grp: 2 — RO Val: 2.8 — SRO Val: 3.2 — 55.43 ☐

System: 215003 — Wide Range Nuetron Monitor (WRNM) System

KA Group Num: A2 — Ability to predict the impact of the following on the system, and based on those predictions, use procedures to correct, control or mitigate the consequences. . . .

KA Detail Num: A2.01 — Power supply degraded.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 20C205L G-3	210 G-3		1	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
WRNM	PLOT5060C	D	16	1	4a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
WRNM Schematic	M-1-S-37		13/32	2/2	

31

Question Data for Test: 2001 SRO

Question: 172

Peach Bottom Unit 2 is operating at rated power with the recirc flow at 90%. A positive reactivity addition occurs which causes an APRM scram. Recirc flow was constant prior to the scram.

Evaluate this transient and determine the reactor power at which the reactor is expected to scram from an APRM signal.

- ☐ A — 114.6% Power
- ☑ B — 117.6% Power
- ☐ C — 119.3% Power
- ☐ D — 123.8% Power

Explanation of Answer

A. Incorrect - APRM Rod Block Setpoint
B. Correct - This would be the "Clamp" setpoint for the STP flow biased APRM scram
C. Incorrect - This is the Fixed High Neutron Flux Scram setpoint
D. Incorrect - Calculated scram setpoint using the flow biased scram formula (.66W + 64.4) without regard for the "Clamp"

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	None

KA Information

Tier: SYS — RO Grp: 1 — SRO Grp: 1 — RO Val: 3.7 — SRO Val: 3.7 — 55.43 ☐

System: 215005 — Average Power Range Monitor/Local Power Range Monitor System

KA Group Num: K4 — Knowledge of the System design feature(s) and/or interlocks which provide for the following.

KA Detail Num: K4.07 — Flow biased trip setpoints.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Exhibit GP-5-1	GP-5		1	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PRNM/OPRM/RBM	PLOT5060	C	23	2	4f

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
APRM HIGH	ARC 211 B-2			4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
APRM/OPRM HI HI/INOP	ARC 211 A-3			5	

32

Question Data for Test: 2001 SRO

Question: 173

During power ascension on Unit 3, control rods are being withdrawn to achieve the target rod pattern when APRM #3 spikes to 121% indicated flux and remains there due to a drawer malfunction. All other APRMs are responding normally.

Which one of the following identifies the necessary action, if any, to continue rod withdrawal?

- [x] A. Bypassing of ARPM #3 is required to clear the rod block to permit continued rod withdrawal.
- [ ] B. Bypassing of APRM #3 is required to clear the rod block, permit resetting the half scram, and continued rod withdrawal.
- [ ] C. Following scram recovery, bypassing of APRM #3 is required to permit resetting the full scram prior to attempting rod withdrawal.
- [ ] D. NO action is required to permit continued rod withdrawal.

Explanation of Answer

A. Correct
B. Incorrect - A half scram will NOT occur
C. Incorrect - A full scram will NOT occur
D. Incorrect - The rod block must be cleared by bypassing the APRM

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.6 | SRO Val: 3.7 | 55.43

System: 215005 — Average Power Range Monitor/Local Power Range Monitor System

KA Group Num: A2 — Ability to predict the impacts of the following on the system; and based on those predictions, use procedures to correct, control, or mitigate the consequences. . . .

KA Detail Num: A2.02 — Upscale or downscale trips.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 211 A-3	211 A-3		1	6	

33

Question Data for Test: 2001 SRO

Question: 175

Peach Bottom Unit 2 was operating at full power when a Group I isolation occurred. The reactor initially failed to scram until all rods were inserted by ARI.

- Reactor pressure spiked to 1275 psig, but is now being controlled at approximately 1040 psig using the SRVs.
- Reactor level initially dropped to -50", but has recovered and is being maintained at approximately 20" using RCIC. HPCI failed due to a governor malfunction.
- Current CST level is 28 feet.
- Primary Containment parameters; Torus temperature = 98 degrees F, Torus Level = 16 ft., Drywell Pressure = 1.6 psig, Drywell Temperature = 143 degrees F.

Under these plant conditions, which of the following statements is correct regarding RCIC CST to CST operations?

RCIC CST-to-CST operation is prevented:

- ☐ A — until the RCIC System automatic initiation signal seal-in is reset.
- ☐ B — until the HPCI System automatic initiation signal seal-in is reset.
- ☐ C — due to the RCIC suction swap to the Torus.
- ☑ D — due to the HPCI suction swap to the Torus.

Explanation of Answer:

A. & B. Initiation signals do not seal-in to the valves.
C. RCIC suction does not swap on Torus high level.
D. HPCI suction swap at 15'10", closing "Cond Tank Return" valve (MO-2-23-24) which must be open for RCIC CST to CST.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.4 | SRO Val: 3.3 | 55.43

System:	217000	Reactor Core Isolation Cooling (RCIC)
KA Group Num:	A2	Ability to (a) predict the impacts of the following on the system; and (b) based on predictions, use procedures to correc, control, or mitigate. . . .
KA Detail Num:	A2.03	Valve closures.

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RCIC System Manual Operation	SO 13.1.B-2	4.4		6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RCIC Lesson Plan	PLOT5013	E.1	28	0	5

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023			1	6

34

Question Data for Test: 2001 SRO

Question: 178

A Refuel Outage is in progress on Peach Bottom Unit 3. The following conditions exist:

- 3C RHR Pump is operating in shutdown cooling.
- Power is lost to the 30Y33 panel due to failure of the Manual Transfer Switch.
- PCIS Shutdown Cooling logic power is lost (due to the loss of 30Y33.)

Which one of the following describes the effect this will have on the operating shutdown cooling loop?

- [x] A — The 3C RHR Pump will trip due to a loss of shutdown cooling suction path.
- [ ] B — The 3C RHR Pump will be damaged due to operating at low flows with no minimum flow protection.
- [ ] C — The RHR Inboard Injection Valve (MO-3-10-25A) will close, the 3C RHR Pump will remain running on minimum flow.
- [ ] D — The Shutdown Cooling Valves will fail "As Is" on the loss of logic power, shutdown cooling will remain in service.

Explanation of Answer:

A. Correct - MO-17 and 18 will close causing the "C" RHR pump to trip on a loss of suction path.
B. Incorrect - The pump will trip.
C. Incorrect - MO-25 will close, but the pump will trip on a loss of suction.
D. Incorrect - MO-17, 18 and 25A will close, pump will trip.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.2 SRO Val: 3.3 55.43

System: 223002 Primary Containment Isolation System

KA Group Num: K3 — Knowledge of the effect that a loss or malfunction of the system will have on the following:

KA Detail Num: K3.16 — Shutdown Cooling System / RHR

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Residual Heat Removal	PLOT5010	II.D.1.b.12	26	1	3.0

Question Data for Test: 2001 SRO

Question: 179

A plant startup is in progress on Peach Bottom Unit 2. The following conditions exist:

- The Rx Mode Switch is in "Startup".
- Two Turbine Bypass Valves are open.
- Reactor pressure is 940 psig and steady.

Which one of the following describes the plant response, if any, when the "PCIS System I Main Steam Line High Flow" pressure transmitter (DPT-2-118A) fails high?

- [x] A. A half Group I Isolation will occur.
- [ ] B. The associated steam lines MSIV will go closed, a reactor scram will occur due to high reactor pressure.
- [ ] C. No effect, this isolation is bypassed with the mode switch out of "Run".
- [ ] D. All the MSIVs will close due to a Group I Isolation.

Explanation of Answer:
A. Correct - One out of two taken twice logic.
B. Incorrect - Half isolation will occur, no MSIVs will close.
C. Incorrect - Half isolation will occur, this Group I signal is not mode switch dependant.
D. Incorrect - Half isolation, no MSIVs will close.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 2.8 | SRO Val: 2.9 | 55.43 ☐

System: 223002 — Primary Containment Isolation

KA Group Num: K6 — Knowledge of the effect that a loss or malfunction of the following will have on the system.

KA Detail Num: K6.06 — Various Process Instrumentation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PCIS	PLOT5007G	II.A.3.C	9	0	5f

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
System I Mn Stm Line Hi Flow	ARC 227 D-1			2	

36

Question Data for Test: 2001 SRO

Question: 180

Which one of the following describes the Residual Heat Removal (RHR) System physical connections to the Torus Spray Header on Peach Bottom Unit 2?

- [x] A Either loop can spray using the single common spray header in the torus.
- [ ] B Either loop can spray using either loop's spray header in the torus.
- [ ] C Each loop can spray using only its associated loop spray header in the torus.
- [ ] D Only the "B" loop can spray using the single spray header in the torus.

Explanation of Answer

A. Correct - Both loops connect to one 100% capacity spray ring in torus.
B. Incorrect - Each loop does not have it's own header.
C. Incorrect - Each loop does not have it's own header.
D. Incorrect - Both loops supply a common spray header.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 1 RO Val: 3.4 SRO Val: 3.6 55.43 ☐

System: 226001 — RHR / LPCI Containment Spray System Mode

KA Group Num: K1 — Knowledge of the physical connections and/or cause - effect relationships between the system and:

KA Detail Num: K1.01 — Suppression Pool

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Sustem P&ID	M-361		1	75	
RHR	PLOT5010	B	14	2	1b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Initiation of Torus Sprays Using RH	T-204	4	1	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

Question Data for Test: 2001 SRO

Question: 181

A Design Bases Loss of Coolant Accident (LOCA) has occurred on Unit 2 while operating at 100% power. Valve logic failures prevent initiation of torus and drywell sprays. All other systems function normally.

Which one of the following identifies the expected drywell temperature response to this event?

Peak drywell bulk average temepratura:

- ☐ A Will not exceed 200 degrees F.
- ☐ B Will not exceed 212 degrees F.
- ☑ C May exceed 281 degrees F.
- ☐ D May exceed 340 degrees F.

Explanation of Answer

A. Incorrect - 200 degrees F is ~2 psig in DW.
B. Incorrect - 212 is the boiling point of water
C. Correct - Without sprays peak DW bulk ave reaches +295 degrees F DBA
D. Incorrect - This is the temperature for a small break LOCA

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 1 RO Val: 3.5 SRO Val: 3.5 55.43 ☐

System: 226001 RHR / LPCI: Containment Spray System Mode

KA Group Num: K3 Knowledge of the effect that a loss of malfunction of the system will have on the following:

KA Detail Num: K3.02 Containment / Drywell / Suppression Chamber Temperature.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control Base	T-102 Bases	DW/T-13	23	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560			8	9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DBA LOCA	PLOT1670	D	7	6	3

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Question Data for Test: 2001 SRO

Question: 182

A Loss of Coolant Accident (LOCA) occurred on Peach Bottom Unit 3. The following conditions existed:

- The 'A' Loop of RHR was blocked
- Reactor Pressure was 850 psig and slowly lowering
- Drywell Pressure was 5 psig and slowly rising
- Wide Range Level Transmitter (LT-72B) indicated downscale
- Fuel Zone Level Transmitter (LT-73B) indicated -225" and steady

One minute later:
- The pressure compensator for Fuel Zone Level (PT-404B) fails low
- You are directed to place Torus Spray in service in accordance with T-203-3 "Initiation of Torus Sprays using RHR".

Given these plant conditions, which of the following actions is required to permit Torus Spray to be lined up with the 'B' Loop of RHR.

- [x] A — Place Switches S18 (CTMT Spray Override 2/3 Core Coverage) in the "ON" position and S17 (CTMT Spray Vlv Cont) in manual
- [ ] B — Depress the S33B (Containment Spray Valve Reset) Pushbutton and place S17 (CTMT Spray Vlv Cont) in manual.
- [ ] C — Place Switch S17 (CTMT Spray Vlv Cont) in the "Manual" position
- [ ] D — Depress the S1B (LPCI Lockout Reset) Pushbutton

Explanation of Answer:

A. Correct - Torus logic will see a lower than actual water level, S18 Switch will override this low level condition, S17 will allow diverting RHR flow from the LPCI mode.
B. Incorrect - The S33B is not used until the LOCA is over
C. Incorrect - A loss of pressure compensation would cause torus spray logic to see a lower than actual level. Level would not meet the 2/3 core coverage interlock.
D. Incorrect - The S1B is used to reset the logic after the LOCA signal is clear

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.1 | SRO Val: 3.2 | 55.43 ☐

System:	230000	RHR/LPCI: Torus Spray Mode
KA Group Num:	K1	Knowledge of the physical connectins and/or cause-effect relationships between the following:
KA Detail Num:	K1.08	Nuclear Boiler Instrumentation

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Residual Heat Removal	PLOT5010	II.D.1.a.3	26	001	1.j

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Vessel Instrumentation	PLOT5002B	II.C.3.b	18	000	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
P&ID Vessel Instrumentation	M-352		6	38	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Elect Schematic Diagram	M-1-S-65		50	95	

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Question Data for Test: 2001 SRO

Question: 185

A Refueling Outage is in progress on Peach Bottom Unit 2. You are the Fuel Handling Director and the following plant conditions exist:

- The Unit is in MODE 5.
- The Mode Selector Switch is in "Refuel"
- The Refueling Platform is in operation over the spent fuel pool
- A fuel bundle has been loaded on the Main Hoist and raised out of the fuel pool storage rack
- A rod block is then received

Evaluate these conditions and determine which of the following actions caused the 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 selected but did NOT withdraw, a single control rod.

Explanation of Answer

A. Incorrect - Would clear the rod block if not over the core.
B. Incorrect - No change in conditions for rod block for this condition alone.
C. Correct - A rod block is expected with any refuel hoist loaded and platform over the core.
D. Incorrect - Input for fuel hoist interlock.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	N/A

KA Information

Tier: SYS RO Grp: 3 SRO Grp: 2 RO Val: 2.6 SRO Val: 3.5 55.43 ☑

System:	234000	Fuel Handling Equipment
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.27	Knowledge of the refueling process.

Question Source Information

Ques Source: 1998 PBAPS NRC Exam

Question Source:

Ques Mod Met: Minor Enhansements

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Receipt of Rod Blocks	SO-62.7.A-2	Attachment 1	5	19	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Refueling Bridge and Platform	NLSRO0762		36	3	13

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Manual Control System	PLOT5062	II.D.2.A	15	1	1g

40

Question Data for Test: 2001 SRO

Question: 186

A reactor startup is in progress on Unit 2 with the reactor critical on Range 2 of the WRNMs. I&C Testing caused multiple channels of the Main Steam Line (MSL) flow instruments to fail upscale causing a full Group I Isolation on high Main Steam Line (MSL) flow. RPS did not actuate. All other systems responded as designed. As the Reactor Operator, determine if an ATWS is in progress and why.

An ATWS condition:

- ☐ A — exists since a scram should have occurred on high MSL flow.
- ☐ B — exists since a scram should have occurred on MSIV closure.
- ☐ C — does NOT exist since the high MSL flow scram is bypassed with the mode switch NOT in run.
- ☑ D — does NOT exist since the MSIV closure scram is bypassed with the mode switch NOT in run.

Explanation of Answer

A. Incorrect - There is no high MSL flow scram.
B. Incorrect - The MSIV closure scram is bypassed with the mode switch not in run.
C. Incorrect - There is no high MSL flow scram.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 3 | RO Val: 3.1 | SRO Val: 3.2 | 55.43 ☐

System:	239001	Main and Reheat Steam System
KA Group Num:	K4	Knowledge of the system design feature(s) and/or interlocks which provide for the following:
KA Detail Num:	K4.05	Steam Flow Measurement

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Steam & Pressure Relief	PLOT5001A		47	0	4.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PCIS	PLOT5007G		15	0	1a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

41

Question Data for Test: 2001 SRO

Question: 187

Peach Bottom Unit 3 is performing a plant startup and is currently holding power at 70% while Condensate Demineralizer work is in progress. Which of the following describes the plant response to a Condensate Demineralizer being returned to service prior to being completely filled and vented?

- [x] A — Main Steam Line Radiation Monitors indication will rise.
- [ ] B — Main Stack Radiation Monitors indication will rise.
- [ ] C — Reactor Feedwater Pumps will trip on low suction pressure.
- [ ] D — Main Condenser Vacuum will degrade.

Explanation of Answer

A. Correct - GE Sil 297 states that rad will rise due to additional N-16 production.
B. Incorrect - N-16 radiation will decay before reaching this location.
C. Incorrect - RFP low suction pressure trips have a time delay preventing a trip with even momentary cavitation.
D. Incorrect - The amount of air in one demin is NOT sufficient to impact condenser vacuum after being disbursed in the reactor vessel.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS — RO Grp: 2 — SRO Grp: 3 — RO Val: 3.6 — SRO Val: 3.6 — 55.43 ☐

System: 239001 — Main and Reheat Steam System

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: A1.05 — Main Steam Line Radiation Monitors

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensate	PLOT5005	II.E.1.C	24	0	5a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
GE SIL	297		1	S1	

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Question Data for Test: 2001 SRO

Question: 188

The following conditions existed at Peach Bottom:

- Both units were operating at full power with no testing in progress.
- Unit 2 Safety Relief Valve "B" has a bellows failure alarm present.

The following event ocurred:

- A small explosion and fast spreading fire erupted when a spark ignited fumes from a can of paint thinner present in the Control Room.
- Both Units were scrammed in accordance with ON-114 "Actual Fire Reported in the Power Block, Diesel Generator Building, Emergency Plan, Inner Screen, or Emergency Cooling Towers". All scram actions were completed.
- The Control Room was evacuated in accordance with SE-10, "Alternative Shutdown".
- The fire has caused the SV-8130A and B, Backup Instrument Nitrogen Valves, to close.

For these conditions, which of the following SRVs are expected to be available for manual pressure control in accordance with SE-10, "Alternative Shutdown"?

- [x] A ONLY the "A" and "B" SRVs are expected to be available.
- [ ] B ONLY the "A" and "K" SRVs are expected to be available.
- [ ] C ONLY the "H" and "E" SRVs are expected to be available.
- [ ] D ONLY the "H" and "L" SRVs are expected to be available.

Explanation of Answer

A. Correct - the "A" and "B" are fire protected and will have a nitrogen supply via a bypass line that is installed in SE-10.
B. Incorrect - Since bellows failure does not impact manual operation of the "B" SRV. Also, the "K" SRV is not expected to be available in a fire requiring SE-10 entry.
C. Incorrect - The "H" and "E" SRVs can be remotely operated but not during these conditons.
D. Incorrect - The "H" and "L" SRVs can be remotely operated but not during these conditons.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS   RO Grp: 1   SRO Grp: 1   RO Val: 3.4   SRO Val: 3.5   55.43

System: 239002   Relief / Safety Valves

KA Group Num:	K6	Knowledge of the effect that a loss or malfunction of the following will have on the system:
KA Detail Num:	K6.02	Air (Nitrogen) Supply

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE-10 Alternative Shutdown	SE-10	Flowchart	1-2	11	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Steam and Pressure Relief	PLOT5001A	II.E.6.f	37-38	1	6i

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE-10 Alternative Shutdown Bases	SE-10	Step ASD/R-13	8	12	

43

Question Data for Test: 2001 SRO

Question: 190

Peach Bottom Unit 2 is operating at 35% power. "Stator Liquid In-Out Hi Temp" AND "Generator Stator Slots Hi Temp" alarms are received. Upon investigation the Plant Reactor Operator determines that an automatic Turbine Generator runback is occurring.

Which of the following describes the appropriate operator response to this condition?

The crew should verify that:

- [ ] A. BOTH Recirculation Pumps trip, scram, and enter T-100, "Scram Condition".
- [ ] B. "Load Set" runs back continuously until generator current is below 7726 amps. If the runback stops before 7726 amps, perform a GP-4, "Manual Reactor Scram".
- [ ] C. "Load Set" runs back until generator current is less than 7726 amps and that bypass valves go full open. Perform GP-9, "Fast Reactor Power Reduction", to prevent a reactor scram.
- [x] D. "Load Set" runs back in pulses until generator current is below 7726 amps and that bypass valves open to control reactor pressure.

Explanation of Answer

A. Incorrect - Recirc trips only occur if initial power is above 45%.
B. Incorrect - The main generator runback will reduce power in pulses to 7726 amps which equates to approximately 23% power. A manual scram is not required unless something does not perform properly.
C. Incorrect - The main generator runback will reduce power in pulses to 7726 amps which equates to approximately 23% power. Bypass valves will only have to absorb approximately 17% power which is well within their capability.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 2.8 | SRO Val: 2.9 | 55.43 ☐

System: 241000 — Reactor / Turbine Pressure Regulating System

KA Group Num: K5 — Knowledge of the oeprational implications of the following concepts as they apply to the system:

KA Detail Num: K5.05 — Turbine Inlet Pressure vs. Turbine Load

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Stator Cooling	OT-113			6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
EHC Logic Lesson Plan	PLOT 5001DL	C.4	19	0	5c

44

Question Data for Test: 2001 SRO

Question: 191

Peach Bottom Unit 2 is operating at full power when the Load Limit Potentiometer output fails causing an output signal of 80%.

Select from the following statements, the one which best represents the plants response to this EHC Logic System failure.

Reactor power and pressure:

- ☐ A — will rise resulting in a reactor scram. Condenser vacuum improves.
- ☐ B — will lower resulting in a Group I Isolation and reactor scram. Condenser vacuum gets worse.
- ☐ C — are stable. Condenser vacuum improves.
- ☑ D — are stable. Condenser vacuum gets worse.

Explanation of Answer: Control valves would close, bypass valves would open maintaining reactor pressure and power relatively stable. Condenser vacuum gets worse because of the increased energy of the steam put in the condenser from the bypass valves.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.4 SRO Val: 3.4 55.43 ☐

System: 241000 — Reactor / Turbine Regulating System

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: A1.21 — Main Condenser Vacuum

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Turbine Startup	SO 1B.1.A-2	4.3	4	26	

45

Question Data for Test: 2001 SRO

Question: 192

Unit 2 is operating in MODE 1 with the following conditions present:

- Main Generator Load is 1100 Mwe.
- Power factor is .95 lagging.
- Generator hydrogen pressure is 60 psig.

The Power System Director contacts you and requests that you raise reactive loading to 380 MVARs. Use the attached generator capability curve to determine if you can meet this request and what the MAXIMUM reactive loading would be under these conditions.

With the current Main Generator loading, the Power System Director's requested reactive loading of 380 MVARs is:

- [ ] A. NOT acceptable. Maximum reactive loading is 220 MVARs.
- [ ] B. NOT acceptable. Maximum reactive loading is 360 MVARs.
- [x] C. acceptable. Maximum reactive loading is 390 MVARs.
- [ ] D. acceptable. Maximum reactive loading is 590 MVARs.

Explanation of Answer:

A. Incorrect - Used .98 Power Factor Line
B. Incorrect - Used leading versus lagging side of curve
C. Correct
D. Incorrect - Used 75 psig hydrogen pressure curve

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	Unit 2 Main Generator Estimated Capability Curve

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.8 SRO Val: 3.1 55.43 ☑

System:	245000	Main Turbine Generator and Auxiliary Systems
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.25	Ability to Interpret Station reference materials such as graphs/nonographs/and tables

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

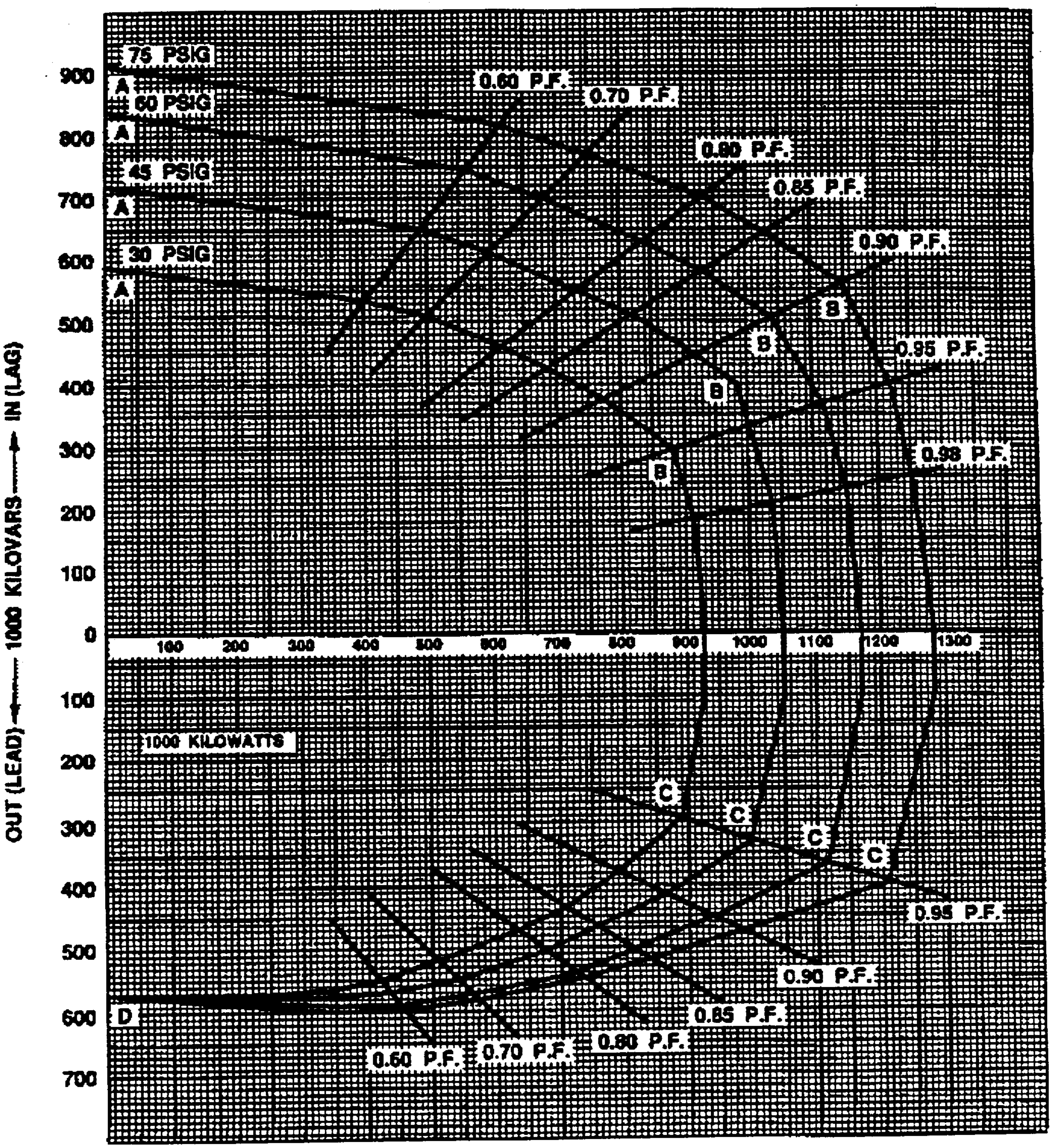
References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Generator Synchronizing and	SO 50.1.A	Figure 1	Last	7	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Generator and Auxiliaries	PLOT5050	II.N.1.a	55	2	10

FIGURE 1

ATB 4 POLE 1,280,000 KVA 1800 RPM 22,000 VOLTS
0.90 P.F. 0.60 SCR 75 PSIG HYDROGEN PRESSURE 500 VOLTS EXCITATION



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
LOWER SECTION OF FIGURE 1 OUT (LEAD) IS UNANALYZED FOR OPERATIONS

Question Data for Test: 2001 SRO

46

Question: 193

Peach Bottom Unit 2 has scrammed due to a loss of feedwater. The following conditions exist:

- Reactor pressue is 700 psig and being lowered in accordance with T-111, "Level Restoration".
- Reactor level is -160" and lowering.
- No high pressure injection is available.
- 'A' Condensate Pump is running.
- All of the RHR Pumps are running.
- 'B' and 'D' Core Spray Pumps are running.
- No reactor coolant system leak exists

When level reaches -172", T-112 "Emergency Blowdown" is performed. Reactor level continues to lower until RPV pressure reaches 320 psig. Reactor level then begins to rise.

Evaluate these conditions and select the answer which explains the plants response to this condition.

- ☐ A — At a reactor pressure of 550 psig, the 'A' Condensate Pump injected. The flow from 5 SRVs exceeds capacity of one condensate pump and reactor level did not rise until the CS Pumps began injecting.
- ☐ B — At a reactor pressure of 550 psig, the 'A' Condensate Pump injected. The flow from 5 SRVs exceeds capacity of one condensate pump and reactor level did not rise until the RHR Pumps began injecting.
- ☑ C — The 'A' Condensate Pump is capable of restoring reactor level with 5 SRVs open and did not inject as normally expected. Reactor level did not rise until the CS pumps began injecting.
- ☐ D — The 'A' Condensate Pump is capable of restoring reactor level with 5 SRVs open and did not inject as normally expected. Reactor level did not rise until the RHR Pumps began injecting.

Explanation of Answer: Condensate pumps are expected to inject when RPV pressure is 600 psig. If condensate was lined up to inject, T-111 allows lowering RPV pressure to inject with condensate before level reaches -172". During T-112, with 5 SRVs open, flow throught the SRVs would be 30% stm flow or less. Condensate pumps are rated for 33 1/3% flow. CS Pumps shut off head 330 psi, RHR Pump shutoff head 305 psi therefore CS injects before RHR during T-112.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	

KA Information

Tier: SYS — RO Grp: 2 — SRO Grp: 3 — RO Val: 3.5 — SRO Val: 3.8 — 55.43 ✓

System: 256000 Reactor Condensate System

KA Group Num:	2.4	Emergency Procedures/Plan
KA Detail Num:	2.4.48	Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensate System LP	PLOT5005			0	3.f

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Level Restoration	T-111	LR6,7		10	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Level Restoration Bases	T-111 Bases	LR6,7		9	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Steam & Press Relief LP	PLOT5001A			01	

47

Question Data for Test: 2001 SRO

Question: 196

A loss of feedwater transient has resulted in an automatic low reactor level scram on Peach Bottom Unit 2.

While verifying automatic actions, the Plant Reactor Operator should expect which of the following Standby Gas Treatment (SBGT) conditions?

- [ ] A — All three fans and both filter trains should be in standby.
- [ ] B — The "A" fan should have auto started and one filter train should be aligned.
- [x] C — The "A" and "B" fans should have auto started and both filter trains should be aligned.
- [ ] D — The "B" and "C" fans should have auto started and both filter trains should be aligned.

Explanation of Answer

A. Incorrect - Level <1" scram occurred.
B. Incorrect - Two trains intitiate and two fans.
C. Correct - Level <1" scram initiates SBGT A and B for Unit 2 and both filter trains align.
D. Incorrect - This would occur on Unit 3 Low Level event.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.7 SRO Val: 3.8 55.43

System: 261000 Standby Gas Treatment System

KA Group Num: K4 — Knowledge of system design feature(s) and/or interlocks which provide for the following:

KA Detail Num: K4.01 — Automatic System Initiation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Low Level	OT-100	4	2	9	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Gas Treatment	PLOT5009A	D		0	4.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Gas Treatment Auto Init.	SO 9A.1.C	4	1	9	

48

Question Data for Test: 2001 SRO

Question: 197

PBAPS Unit 2 is operating at 100% power with the electric plant in a normal lineup when the SU-25 breaker trips on low SF6 pressure.

Select the response below which describes the effects on the SU-25 breaker trip on PBAPS Unit 2.

A fast transfer to their alternate sources will occur for 4KV busses:

- [x] A — E12 and E32. A Group II inboard half isolation will be received.
- [ ] B — E22 and E42. A Group II outboard half isolation will be received.
- [ ] C — E12 and E32. The E1 and E3 Emergency Diesels start and run unloaded; a Group II inboard half isolation will be received.
- [ ] D — E22 and E42. The E2 and E4 Emergency Diesels start and run unloaded; a Group II outboard half isolation will be received.

Explanation of Answer:

E22 and E42 normally supplied by 3EA and 343SU, E12 and E32 from SU-2.
DG do not start unless the fast transfer to alternate fails.
Group II outboard half isolation occurs with loss of power to 20Y34, 20Y034 is power from E22.
Group II inboard half isolation occurs with loss of power to 20Y33, 20Y33 is powered from E12.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 1 | RO Val: 3.1 | SRO Val: 3.4 | 55.43 ☐

System: 262001 — A.C. Electrical Distribution

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: A1.01 — Effect on instrumentation and controls of switching power supplies.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
4KV Distribution LP	PLOT5054	D.7,E.1	23,26	2	3g

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
kKV Distribution LP	PLOT5054	D.7,E.1	23,26	2	6

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
GP1, II & III Inbod Half Isol.	GP-8C	2	2	17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
GP I, II & III Outbd Half Isol.	GP-8D	2	2	12	

49

Question Data for Test: 2001 SRO

Question: 198

A complete loss of offsite power has occurred at Peach Bottom. No Diesel Generators are available. To minimize the battery discharge rate, SE-11 "Loss of Off Site Power" directs performance of SE-11 Attachment T, "DC Load Shed".

Completion of Attachment T results in deenergization of which of the following circuits?

- [x] A Alternate Rod Insertion (ARI) Logic
- [ ] B Reactor Core Isolation Cooling (RCIC) Logic
- [ ] C Emergency Core Cooling System (ECCS) Logic
- [ ] D Safety Relief Valve (SRV) Control

Explanation of Answer

A. Correct - ARI is deenergized by Attachment T.
B. Incorrect - Are specifically listed in SE-11 Bases as loads to which Attachment T will maintain power.
C. Incorrect - Are specifically listed in SE-11 Bases as loads to which Attachment T will maintain power.
D. Incorrect - Are specifically listed in SE-11 Bases as loads to which Attachment T will maintain power.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.5 SRO Val: 2.8 55.43 ☐

System: 263000 D.C. Electrical Distribution

KA Group Num: A1 Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: A1.01 Battery Charging/Discharging Rate

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Off Site Power	SE-11 Bases	Sheet 5	49	11	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DC Load Shed	SE-11 Att.			8	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE Procedures Lesson Plan	PLOT1555			5	9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE Procedures Lesson Plan	PLOT1555			5	13

50

Question Data for Test: 2001 SRO

Question: 199

While operating at rated conditions with a normal electrical lineup, a piece of scaffolding inadvertently strikes the normal off-site feeder breaker (E-212) for the E-12 bus. This results in an E-212 breaker trip.

The E-1 Emergency Diesel Generator will:

- [ ] A — auto start and reenergize the bus.
- [ ] B — auto start and the bus will reenergize from the alternate feeder breaker.
- [ ] C — NOT auto start and the bus will remain deenergized.
- [x] D — NOT auto start and the alternate feeder breaker will reenergize the bus.

Explanation of Answer

A. Incorrect - Only if alternate breaker failed to reenergize the bus.
B. Incorrect - No auto start because a fast transfer will occur.
C. Incorrect - Fast transfer will occur and reenergize the bus.
D. Correct - Alternate breaker closure will reenergize the bus. Diesel will not get a start signal.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.8 SRO Val: 4.1 55.43 ☐

System: 264000 Emergency Generators

KA Group Num: K1 Knowledge of the physical connections and/or cause effect relationships between.

KA Detail Num: K1.01 A.C. Electrical Distribution

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DG Auto Start and Loading	SO 54.7.E	4		5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Diesel Generators	PLOT5052	D	12/38	0	1.a

51

Question Data for Test: 2001 SRO

Question: 200

The following conditions exist:

- The E-22 4KV Bus has lost power.
- The fast transfer to its alternate off-site source failed
- The E-2 Diesel Generator (DG) started automatically and loaded the E-22 4KV Bus.

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

The E-2 DG is operating in:

- [ ] A. Droop (Parallel), the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes or it will return to the original mode.
- [ ] B. Droop (Parallel), the DG Auto Start Bypass pushbutton must be pressed and synch may be completed without concern for it returning to the original mode.
- [ ] C. Isochronous (Unit), the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes or it will return to the original mode.
- [x] D. Isochronous (Unit), the DG Auto Start Bypass pushbutton must be pressed and synch may be completed without concern for it returning to the original mode.

Explanation of Answer

A. Incorrect - Will be in the Unit Mode
B. Incorrect - Will be in the Unit Mode
C. Incorrect - Without a MCA signal, it will not transfer back to the original mode.
D. Correct answer.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.7 | SRO Val: 3.7 | 55.43 ☐

System:	264000	Emergency Generators (Diesel/Jet)
KA Group Num:	A4	Ability to manually operate and/or monitor in the control room.
KA Detail Num:	A4.04	Manual start, loading, and stopping of emergency generator: Plant Specific

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Diesel Generators and Auxiliaries	PLOT5052	7	45	0	4

Question Data for Test: 2001 SRO

Question: 201

Peach Bottom Unit 2 was operating at full power when it experienced a recombiner transient. The PRO reports that he believes that the cause is "Recombiner Process Flashback" to the SJAE after condenser.

Select the following indication that would support this diagnosis.

- [ ] A. A drop in Main Condenser Vacuum (vacuum degrading) due to excess hydrogen and oxygen in the Main Condenser.
- [x] B. A rise in Air Ejector Discharge Radiation levels on RR-2-17-152 due to a drop in dilution flow.
- [ ] C. A drop in Adsorber Inlet flow on FR-4020 due to reduced offgas flow.
- [ ] D. A rise in Recombiner Delta T on DTR-4025 due to excess hydrogen and oxygen present in the recombiner.

Explanation of Answer:
A. Incorrect - As long as recombination is occurring vacuum will remain steady.
B. Correct - The recombination of H2 & O2 in the after condenser causes less dilution flow at the radiation monitor.
C. Incorrect - Adsorber Inlet flow is based on air inleakage and air inleakage remains steady.
D. Incorrect - The hydrogen and oxygen concentrations have not gone up.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.1 | SRO Val: 3.1 | 55.43 ☐

System: 271000 — Offgas System

KA Group Num: K1 — Knowledge of the physical connections and/or cause-effect relationships between the offgas system and the following:

KA Detail Num: K1.01 — Condenser air removal system.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Air Ejector Discharge Radiation Hig	ARC 218 E-2	Cause	1 of 1	7	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Offgas Recombiner System	PLOT5008	II.E.5	33	001	1a

53

Question Data for Test: 2001 SRO

Question: 202

Peach Bottom Unit 3 is operating at 75% power when it experiences a lowering condenser vacuum. The PRO notes that Off Gas flow is below normal and continues to lower.

Diagnose the potential cause of this lowering Off Gas flow.

- [ ] A — Loss of Main Turbine Steam seal pressure
- [ ] B — Leak in the standby Main Feedwater Pump Recirc Line.
- [ ] C — Loss of Steam Packing Exhauster loop seals.
- [x] D — Loss of Recombiner Jet Compressor steam supply pressure.

Explanation of Answer

A. Incorrect - This will result in air inleakage and a rise in offgas flow.
B. Incorrect - This will result in air inleakage and a rise in offgas flow.
C. Incorrect - This will result in air inleakage thru loop seals and a rise in offgas flow.
D. Correct - This will result in lower suction thru sys. and lower flow.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS   RO Grp: 2   SRO Grp: 2   RO Val: 3.2   SRO Val: 3.3   55.43

System: 271000 Offgas System

KA Group Num: K6 — Knowledge of the effect that a loss or malfunction of the following will have on the system:

KA Detail Num: K6.11 — Condenser Vacuum

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensor Low Vacuum Bases	OT-106	Bases	3	18	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
P&ID Offgas Recombiner Sys.	M-331		1	68	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Offgas Recombiner ER Sys.	PLOT5008	E	26	1	6.k

.54

Question Data for Test: 2001 SRO

Question: 204

The Hydrogen Water Chemistry System uses heat detectors to monitor each of the seven shrouded areas per unit.

These heat detectors are necessary because:

- [ ] A ventilation flow through the shroud makes detection of a fire impossible by any other means.
- [x] B hydrogen burns with an invisible flame and would not be discovered by visual inspection.
- [ ] C heat detection can be used to make an early determination of a hydrogen leak before a fire can occur.
- [ ] D heat detection is the only method available for determining a fire in a shroud.

Explanation of Answer

A. Incorrect - Ventilation is limited in the shroud.
B. Correct
C. Incorrect - Once heat is present a fire has already started.
D. Incorrect - The LEXAN covers will turn black in the presence of heat.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.6 SRO Val: 2.7 55.43 ☐

System:	286000	Fire Protection System
KA Group Num:	K5	Knowledge of the operational implications of the following concepts as they apply to Fire Protection System.
KA Detail Num:	K5.06	Heat Detection

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Hydrogen Water Chemistry Ssytem	PLOT5015	II.D.1.I	24	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC Hydrogen Heat Detection	ARC 230 D-2			0	

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Question Data for Test: 2001 SRO

Question: 206

T-103, "Secondary Containment Control", has been entered on Peach Bottom Unit 2. The following conditions exist:

- The reactor is at rated power and pressure.
- Reactor level is 23 inches.
- Reactor Bldg. 165' General Area radiation level is 9000 MR/HR on ARM #2.11.
- Reactor Bldg. 165' General Area temperature is 150 degrees F on TRS-2-13-139 Point #22.
- Torus Room temperature is 117 degrees F on TRS-2-13-139 Point #'s 8, 9 and 15.
- Annunciator 215 E-2, "REAC BLDG FLOOR DRAIN SUMP HI-HI LEVEL" is annunciating.
- All parameters are rising slowly.

In accordance with T-103, peform a:

- ☐ A. GP-3 "Normal Plant Shutdown"
- ☑ B. GP-4 "Manual Reactor Scram"
- ☐ C. GP-9 "Fast Power Reduction"
- ☐ D. T-112 "Emergency Blowdown"

Explanation of Answer:

A. Incorrect - Indication of a primary system leak exists.
B. Correct
C. Incorrect - a primary breach exists
D. Incorrect - The same parameter has not exceeded an action level in more than one area.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	T-103 - SCC-4 thru 11 and Tables R1 and T-3.

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 1 RO Val: 3.3 SRO Val: 3.4 55.43 ☐

System:	290001	Secondary Containment
KA Group Num:	A4	Ability to manually operate and/or monitor in the Control Room:
KA Detail Num:	A4.02	Reactor Building Area Temperatures

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Transient Response Implementatio	PLOT1560	II.C	10	008	5

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Control Ba	T-103 Bases	SCC-7	12	12	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Control	T-103 Flowchart	SCC-7	1	13	

T-103 SH 1 OF 1

SCC SECONDARY CONTAINMENT CONTROL

PEACH BOTTOM ATOMIC POWER STATION TRIP PROCEDURE

REV. NO. 14 | DATE 10/20/00

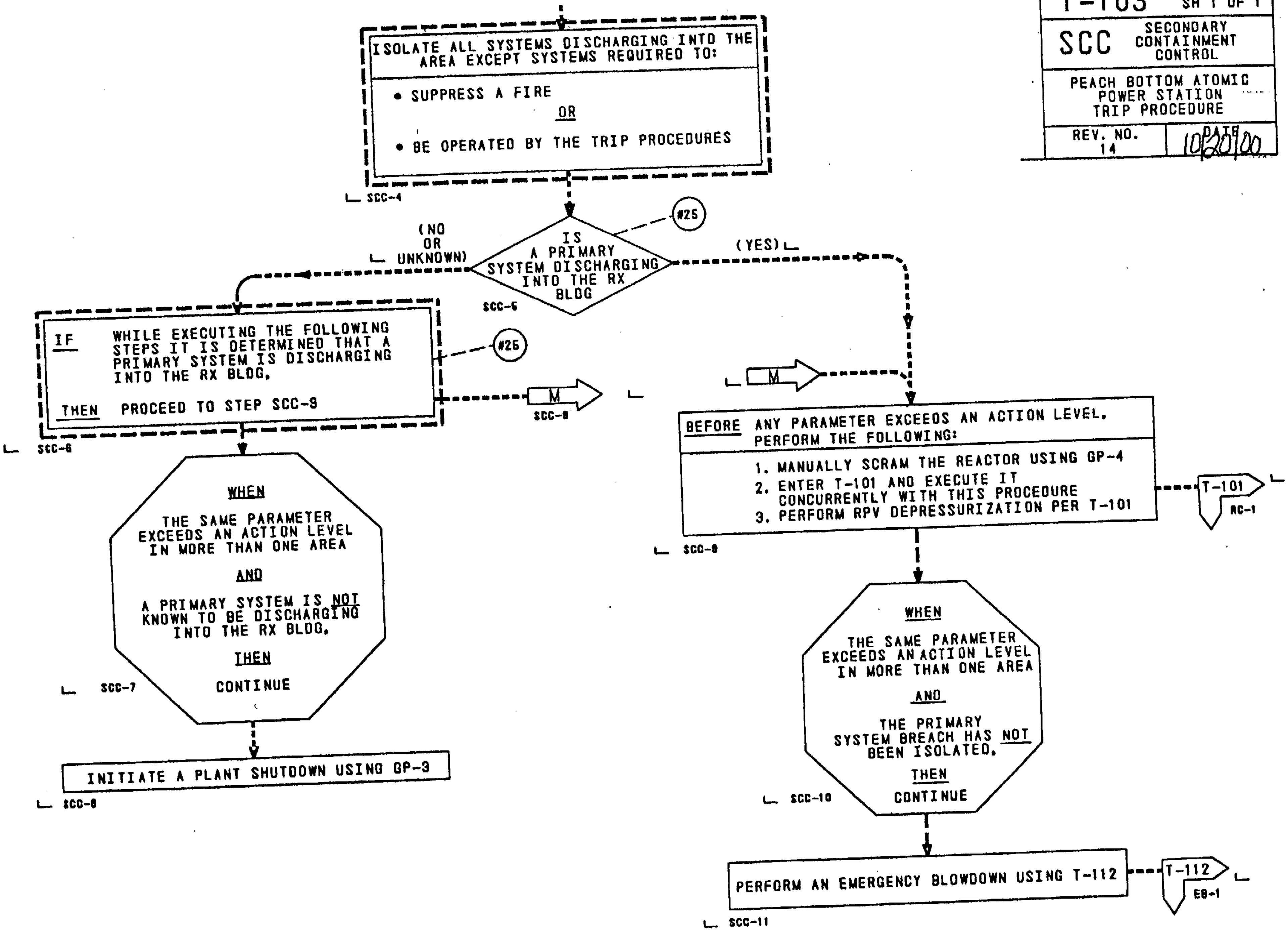

TABLE SC/R-1
RADIATION-ALARM AND ACTION LEVELS

AREA	ALARM LEVEL	ACTION LEVEL (MR/HR)	ARM NUMBER		STATUS
			UNIT 2	UNIT 3	
TORUS ROOM	ALARM SETPT	9×10^3	1.3	5.4	
SUMP ROOM	ALARM SETPT	9×10^3	1.2	5.3	
OR RCIC ROOM	ALARM SETPT	9×10^3	1.5	5.7	
OR HPCI ROOM	ALARM SETPT	9×10^3	1.4	5.6	
A OR C RHR ROOM	ALARM SETPT	9×10^3	1.7	5.9	
B OR D RHR ROOM	ALARM SETPT	9×10^3	1.6	5.8	
A OR C CS ROOM 91'6"/116' EL	ALARM SETPT	9×10^3	1.10	5.10 OR 5.12	
B OR D CS ROOM 91'6"/116' EL	ALARM SETPT	9×10^3	1.8 OR 1.11	6.1	
TIP ROOM	ALARM SETPT	NO ACTION LEVEL	2.8	6.10	
GENERAL AREA 135' EL	ALARM SETPT	9×10^3	2.5, 2.6 OR 2.7	6.7, 6.8 OR 6.9	
RWCU/ISOL VALVE PIT AREA 165' EL	ALARM SETPT	9×10^3	2.12	7.2	
GENERAL AREA 165' EL	ALARM SETPT	9×10^3	2.11	7.1	
GENERAL AREA 195' EL	ALARM SETPT	9×10^3	3.6	7.8	
REFUEL FLOOR	ALARM SETPT	NO ACTION LEVEL	3.7, 3.8, 3.9 OR 3.10	7.9, 7.10, 7.11 OR 7.12	

TABLE SC/T-3

TEMPERATURE-ALARM AND ACTION LEVELS

AREA	ALARM LEVEL (°F)	ACTION LEVEL (°F)	INSTRUMENT TRS-2(3)-13-139 PT # (UNLESS SPECIFIED OTHERWISE)	STATUS
TORUS ROOM	115	135	PT 8,9,14,15,20, OR 24	
RCIC ROOM OR HPCI ROOM	110 110	135 150	PT 2 PT 3	
A RHR ROOM OR C RHR ROOM	110 110	135 135	PT 17 PT 29	
B RHR ROOM OR D RHR ROOM	110 110	135 135	PT 23 PT 6	
A CS ROOM OR C CS ROOM	110 110	135 135	TI-2(3)501 PT 151 TI-2(3)501 PT 152	
B CS ROOM OR D CS ROOM	110 110	135 135	TI-2(3)501 PT 153 TI-2(3)501 PT 154	
STEAM TUNNEL	175	190	PT 1 OR 16	
A ISOL VALVE ROOM (SOUTH)	165	190	PT 12	
B ISOL VALVE ROOM (NORTH)	165	190	PT 18 OR 21	
ISOL VALVE PIT 165' EL	140	150	PT 30	
RWCU REGEN HX ROOM OR A NON REGEN HX ROOM OR B NON REGEN HX ROOM OR A OR B RWCU FLTR DEMIN ROOM OR RWCU BACKWASH VALVE ROOM	160 130 130 115 105	NO ACTION LEVEL	PT 11 PT 28 PT 5 PT 10 OR 27 PT 4	
GENERAL AREA 165' EL (MAY AFFECT RPV LEVEL INST)	105	135	PT 22	

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Question Data for Test: 2001 SRO

Question: 207

Peach Bottom Unit 2 is in MODE 5. Control Rod Drive (CRD) removal and replacement is in progress.

The 'A' Standby Gas Treatment (SGT) Fan has been INOPERABLE for 3 days. It has just been determined that the 'B' SGT Filter Train is INOPERABLE due to an outlet damper failed closed.

Use the attached Technical Specifications to select the answer which describes the required actions for Unit 2.

- ☐ A — Immediately initiate action to suspend operations with the potential to drain the reactor vessel.
- ☐ B — Immediately place the 'B' SGT Fan and 'A' SGT Filter Train in service.
- ☑ C — Restore the 'A' SGT Fan and 'B' SGT Filter Train to OPERABLE within the next 4 days.
- ☐ D — Enter Technical Specification LCO 3.0.3.

Explanation of Answer:
- 7 day clock continues from initial INOPERABILITY
- 1 Subsystem remains operable with 'A' Filter Train and the 'B' Fan.
- TSA 3.6.4.3.A still applies.

Exam Level	Cognitive Level	Facility	Materials
SRO	Application	PBAPS	PB Unit 2 Tech Spec 3.6.4.3

KA Information

Tier: SYS   RO Grp: 2   SRO Grp: 1   RO Val: 2.6   SRO Val: 3.8   55.43 ☑

System: 290001 — Secondary Containment

KA Group Num: 2.2 — Equipment Control

KA Detail Num: 2.2.23 — Ability to track limiting conditions for operation.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PB Tech Spec and Bases		3.6.4.3			

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Lesson Pl	PLOT5009	F,G			7

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment	PLOT5009	F,G			8

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	 Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1	Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
		<u>AND</u>	
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
		<u>AND</u>	
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	--------NOTE--------- LCO 3.0.3 is not applicable. ---------------------	
		Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 15 minutes with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months

Question Data for Test: 2001 SRO

Question: 208

Peach Bottom Unit 2 is operating at full power with the Backup Air Compressor is blocked for maintenance. The "B" Instrument Air Compressor trips due to an electrical failure in the compressor motor.

Under these conditions, a complete loss of instrument air would occur upon the loss of:

- [ ] A ONLY the #1 Aux Bus.
- [ ] B ONLY the #2 Aux Bus.
- [x] C BOTH the #1 and #2 Aux Busses.
- [ ] D BOTH the E-134 and E-324 Busses.

Explanation of Answer

A. Incorrect - Would only trip the "A" air compressor.
B. Incorrect - Would only trip the "C" air compressor.
C. Correct - Would trip both the "A" and "C" air compressors.
D. Incorrect - The Backup Air Compressor is already blocked.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.8 SRO Val: 2.8 55.43 [ ]

System:	300000	Instrument Air System
KA Group Num:	K2	Knowledge of electrical power supplies to following:
KA Detail Num:	K2.01	Instrument Air Compressors

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Compressed Air System	PLOT5036	II.C.1.d	9	0	2

Question Data for Test: 2001 SRO

Question: 209

A trip of the "A" Reactor Recirculation Pump has resulted in entry into Region 2 of the Peach Bottom Unit 2 Power/Flow map.

Which one of the following indications would require a manual scram in accordance with OT-112, "Unexpected/Unexplained Change in Core Flow"?

- [ ] A Greater than a 10% difference between any two APRMs.
- [x] B Greater than a 10% difference peak to peak on any APRM.
- [ ] C LPRM flux noise level rises from 2% to 3%.
- [ ] D OPRM trip setpoint exceeded on any single APRM.

Explanation of Answer

A. Incorrect - Similar to 10% diff between any two APRM flow values.
B. Correct - OT-112 THI Indication (2nd bullet).
C. Incorrect - Flux noise must increase by two or more times
D. Incorrect - Requires a trip of two channels of the OPRMs

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.3 SRO Val: 3.4 55.43 ☐

System: 295001 Partial or complete loss of forced core flow circulation.

KA Group Num: AA1 Ability to operate and/or monitor the following as they apply to:

KA Detail Num: AA1.06 Neutron Monitoring System

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Unexpected/Unexplained Change i	OT-112	2	1	31	
Recirc/Recirc Flow Control	PLOT5002	J	65		1.b

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Question Data for Test: 2001 SRO

Question: 212

PBAPS Unit 2 is operating at 100% power with all equipment operating normally.

The following indications occur simultaneously:

- 2B and 2C Circulating Water Pump Motor amps = 0.
- 2B and 2C Circulating Water Pump Breakers indicate closed.
- 2B and 2C Circulating Water Pump Discharge Valve position indication is not energized.
- Main Condenser Vacuum is dropping (Getting worse).

The CRS directs an immediate plant shutdown due to the impact of the loss of the:

- ☑ A — 2G4 Generator Area Load Center.
- ☐ B — 20D021 125VDC Distribution Panel.
- ☐ C — 20Y050 Uninterruptable AC Distribution Panel.
- ☐ D — E224-P-A Pump Structure Motor Contol Center.

Explanation of Answer: 2B and 2C Circulating Water Pump excitation and discharge value operators are powered from 2G4. Loss of excitation causes pumps to stop leaving 1 CW pump operating. This CW Pump will short cycle through the open discharge valves of the other two pumps. Main Condenser Vac decreases.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	

KA Information

Tier: E/APE — RO Grp: 2 — SRO Grp: 1 — RO Val: 3.4 — SRO Val: 3.7 — 55.43 ☑

System:	295003	Partial or complete loss of AC Power.
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.01	Cause of partial or complete loss of AC power.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
MCC Electrical One Line	E-1602			32	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
13KV and 480V Aux Power Dist. L	PLOT5053			1	3

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Question Data for Test: 2001 SRO

Question: 213

The following conditions exist at Peach Bottom:

- A loss of all off-site power has occurred.
- The Emergency Diesel Generators (DG) are supplying their respective 4KV switchgear.
- 10 minutes later a failure of the 2A Battery (2AD01) results in the loss of 125VDC to 20D21 supplying the E-1 DG.

Which of the following describes the expected status of the DG for this failure?

- [ ] A. The DG will shift to the DROOP (Parallel) mode causing output frequency to drop about 5% to 57 hertz.
- [ ] B. The DG engine will trip on mechanical overspeed due to loss of power to the electrical govenor.
- [x] C. The DG will continue to run at the previous speed and loading.
- [ ] D. The DG voltage will lower due to loss of the exciter field flash supply.

Explanation of Answer: If the D/G is running, all alarms, auxiliary pump starts and automatic trips are lost. However, D/G will not trip due to loss of field and will continue to carry load.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.0 SRO Val: 3.1 55.43 ☐

System:	295004	Partial or complete loss of DC power.
KA Group Num:	AK2	Knowledge of the interrelations between - and the following:
KA Detail Num:	AK2.02	Batteries

Question Source Information

Ques Source: 1997 PBAPS NRC Exam Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Diesel Generator LP	PLOT5052	D.8.C	55	0	6.g

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Question Data for Test: 2001 SRO

Question: 215

Peach Bottom Unit 2 was operating at full power when an earthquake occurred. The following failures resulted:
-The Main Turbine (MT) First Stage Pressure instrument lines failed causing all of the MT First Stage Pressure Switches (PS-2-5-14A thru D) to receive a 0 pressure input.
-A failure of the MT Lube Oil System resulted in a Main Turbine trip two minutes later.

As the assigned Senior Reactor Operator (SRO), you have been directed to conduct a GP-18, "Scram Review", to determine if the plant responded as expected to the transient. Which of the following descibes the plant's expected response to these conditions.

- ☐ A — A Turbine Stop Valve Closure scram should have occured and the EOC-RPT breakers should have tripped.
- ☑ B — The Turbine Stop Valve Closure scram and EOC-RPT trip were bypassed. The reactor should have scramed on high pressure.
- ☐ C — A Turbine Stop Valve Closure scram should have occured. The EOC-RPT trip was bypassed.
- ☐ D — The Turbine Stop Valve Closure scram was bypassed. The EOC-RPT breakers should have tripped. The reactor should have scramed on high pressure.

Explanation of Answer: The TSV closure scram and EOC-RPT Breaker trip are bypassed when PS-2-5-14A thru D sense First Stage Pressure below 138 psig. With all 4 pressure switches sensing 0 psig, both functions are bypassed and the reactor pressure rises to the scram setpoint.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 2 RO Val: 3.7 SRO Val: 3.8 55.43 ☑

System:	295005	Main Turbine Generator Trip
KA Group Num:	AA2.04	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.04	Reactor Pressure

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPS Lesson Plan	PLOT5060F	C.3.b		1	1.l

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc System Lesson Plan	PLOT5002	E.3	34	1	4.m

62

Question Data for Test: 2001 SRO

Question: 216

Peach Bottom Unit 3 was operating at full power when a spurious Group I Isolation occurred.

Select the statement which describes the expected response of the Reactor Feedpumps during the first minute following this transient.

The Reactor Feedpumps will be:

- ☐ A — providing minimum flow into the reactor due to the reactor level swell caused by the Recirc Pump trips.
- ☐ B — providing minimum flow into the reactor due to the high reactor pressure condition caused by the Group I Isolation.
- ☐ C — attempting to provide maximum flow into the reactor, but will not have sufficient steam to operate due to the Group I Isolation.
- ☑ D — providing maximum flow into the reactor due to the reactor level shrink caused by the reactor power drop.

Explanation of Answer:

A. Incorrect - The Recirc Pumps don't automatically trip on a Group I Isolation.
B. Incorrect - The shutoff head of the RFPs is above the SRV lift points.
C. Incorrect - There is sufficient steam to run the RFPs for several minutes following a scram.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE — RO Grp: 1 — SRO Grp: 1 — RO Val: 3.8 — SRO Val: 3.8 — 55.43 ☐

System:	295006	Scram
KA Group Num:	AK2	Knowledge of the interrelations between scram and the following:
KA Detail Num:	AK2.02	Reactor Water Level Control System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
OSPS Reactor Operator Response	NOM-P-10.2:5			0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Chapt	PLOT1527	IIB.5	12	0	1q

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Question Data for Test: 2001 SRO

Question: 218

Following a Peach Bottom Unit 2 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".

The CRS needs this information to determine if:

- ☐ A — reactor level should be lowered to < -60".
- ☐ B — Standby Liquid Control injection is required
- ☑ C — the reactor is shutdown and will remain shutdown during the ensuing cooldown.
- ☐ D — entry into T-117 "Level/Power Control" and exit of T-101 "RPV Control" is required

Explanation of Answer

A. Incorrect - Not required with APRM downscales present
B. Incorrect - SBLC is injected if power > 3% or unknown and the reactor is not shutdown.
C. Correct - Maximum Subcritical Banked Withdrawal Position.
D. Incorrect - T-117 entry required if rods are not inserted, but T-101 is not exited

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 4.3 SRO Val: 4.4 55.43 ☑

System: 295006 SCRAM

KA Group Num: AA2 — Ability to determine and/or interpret the following as they apply to:

KA Detail Num: AA2.02 — Control Rod Position

Question Source Information

Ques Source: 1998 PBAPS NRC Exam

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip Curves, Tables & Limits - Bas		4	3	2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Transient Response Implementatio	PLOT1560			7	6

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Question Data for Test: 2001 SRO

Question: 219

Peach Bottom Unit 2 is being started up in accordance with GP-2, "Normal Plant Startup". The startup has been completed to the point of reactor pressure at 450 psig with 3 Bypass Valves open.

Holding reactor pressure at 450 psig ensures that:

- [ ] A — a sufficient warmup of the feedwater nozzles minimizes the chance of thermal stress cracking.
- [ ] B — the RPV does not exceed 20 degrees F temperature change in a 15 minute interval which corresponds to the administrative limit of 80 degrees F/hr.
- [ ] C — turbine shell warming is monitored and adjusted to maintain turbine first stage pressure below 100 psig.
- [x] D — a reactor feedpump will be operating prior to the reactor pressure exceeding the condensate pump shutoff head.

Explanation of Answer: GP-2, NOTE:
"Reactor pressure is held at 450 psig to ensure that a reactor feedpump is operating prior to reactor pressure exceeding the condensate pump shutoff head".

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 2.9 SRO Val: 3.2 55.43 ☐

System: 295007 High Reactor Pressure

KA Group Num: AK1 — Knowledge of the operational implications of the following concepts as they apply to:

KA Detail Num: AK1.01 — Pump Shutoff Head

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Normal Plant Startup Proc	GP-2	Note	65	96	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
General Plant Procedures LP	PLOT1530	B	7	11	3

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Question Data for Test: 2001 SRO

Question: 220

T-101, "RPV Control", Step RC/P-13, provides a list of systems to be used, as necessary, to stabilize RPV pressure below 1050 psig. The list includes the use of HPCI.

Which of the following conditions would prevent the use of HPCI in the "CST to CST Mode" for RPV pressure control?

- ☐ A. Condensate Storage Tank (CST) level indicates 8 feet.
- ☐ B. Torus water level indicates 15 feet.
- ☐ C. Reactor pressure indicates 150 psig.
- ☑ D. Reactor water level indicates -51 inches.

Explanation of Answer:

A. Incorrect - Swap to Torus is at 5' 7"
B. Incorrect - Swap to Torus is at 15' 6"
C. Incorrect - Steam pressure isolation is at 75 psig.
D. Correct - Initiation signal at -48 inches RPV level, CST-CST prevented with any initiation signal.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.7 SRO Val: 3.8 55.43 ☐

System: 295007 High Reactor Pressure

KA Group Num: AK3 Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.02 HPCI Operation: Plant Specific

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control Bases	T-101B	RC/P-13	24	21	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023	E.1	27	1	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023	C.6 & 7	17	1	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023	C.6 & 7	17	1	6

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Question Data for Test: 2001 SRO

Question: 221

Peach Bottom procedure OT-110, "Reactor High Level", requires that for an unexpected rise in level above +46 inches, the operator is to verify that the Main Turbine is tripped.

Under these conditions, OT-110 requires the main turbine trip to be verified to:

- [ ] A reduce the steaming rate to mimize the effects of moisture carryover with the steam.
- [ ] B ensure that all moisture carryover is routed directly to the main condenser.
- [x] C minimize the risk of turbine damage due to moisture carryover with the steam.
- [ ] D eliminate the need to close the MSIVs if reactor water level reaches the bottom of the steam lines.

Explanation of Answer: T.S. Bases states purpose of HILV/Turbine trip is to prevent turbine damage.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.0 | SRO Val: 3.2 | 55.43 ☐

System: 295008 — High Reactor Water Level

KA Group Num: AK1 — Knowledge of the operational implications of the following concepts as they apply to:

KA Detail Num: AK1.01 — Moisture Carryover

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor High Level Bases	OT-110	3.5	6	6	
Reactor Vessel Lesson Plan	PLOT5004	E.2	38	0	3.d

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Technical Specifications Bases		B.3.3.2.2	3.3-58		

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Question Data for Test: 2001 SRO

Question: 222

You are the CRS while a Peach Bottom Unit 2 Startup is in progress. The following conditions exist:
-The reactor is critical
-Recirc pump suction temperature is 180 degrees F with heatup rate being maintained at 75 deg. F/hr.
-The Reactor Water Clean Up (RWCU) System is in service with two Filter/Demins (F/D) in service and one RWCU recirc pump running.
-CV-2-12-055, "RWCU Dump Flow" valve is open with flow to the main condenser at 150 gpm.
-RPV level is +23 in.

A pressure transmitter fails causing the "CLEAN-UP DRAIN HEADER HI-LO PRESSURE" alarm to activate and the automatic closure of CV-2-12-055.

Select the answer which describes the expected plant response and direction that should be given.

- [ ] A Non-Regenerative Heat Exchanger Outlet Temperature will rise. Direct the PRO to manually secure the RWCU System to prevent an automatic isolation.
- [x] B Reactor level will begin to rise. Direct the Startup Reactor Operator to stop the power ascension until dump flow can be restored.
- [ ] C The core thermal power calculation will give an artificially higher value. Direct the Reactor Engineers to manually insert a substitute value into the process computer.
- [ ] D The "CLEANUP FILTER DEMIN SYSTEM TROUBLE" alarm activates. Direct the PRO to verify that the RWCU pumps have automatically tripped on low flow.

Explanation of Answer

A. Incorrect - NRHX Outlet temp will decrease when dump flow goes down.
B. Correct
C. Incorrect - Per SO 12.1.A, this occurs when dump flow is started.
D. Incorrect - The RWCU pumps no longer automatically trip on low flow.

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.1 SRO Val: 3.3 55.43 ☑

System:	295008	High Reactor Water Level
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.04	Heatup Rate

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met	

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Plant Statup Procedure	GP-2	6.2		95	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RWCU System Startup Proc.	SO 12.1.A-2	3.3, Table 1		27	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RWCU System Lesson Plan	PLOT5012	E.3	18	1	3.b

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Question Data for Test: 2001 SRO

Question: 223

Peach Bottom Unit 3 is operating at full power when a low level transient occurs. The level transient causes a recirculation runback, reactor scram, and ultimately trips the recirc pumps.

Wide range level instruments would indicate ________(1)_______ actual level before the transient and _______(2)_______ actual level after the runback, scram, and pump trip.

- [ ] A — lower than (1), higher than (2)
- [x] B — lower than (1), the same as (2)
- [ ] C — the same as (1), higher than (2)
- [ ] D — higher than (1), lower than (2)

Explanation of Answer

A. Incorrect - Wrong combinations.
B. Correct - WR is calibrated for no recirculation pumps in operation. This results in lower than actual with pumps running. WR is accurate with no recirc flow.
C. Incorrect - Wrong combinations.
D. Incorrect - Wrong combinations.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.9 SRO Val: 4.0 55.43 ☐

System:	295009	Low Reactor Water Level
KA Group Num:	AK2	Knowledge of the interrelations between - and the following:
KA Detail Num:	AK2.01	Reactor Water Level Indication

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT5002	E	38	2	4.o

69

Question Data for Test: 2001 SRO

Question: 224

T-102, "Primary Containment Control" procedure directs containment venting if pressure is expected to exceed PCPL-A (60 psig).

The bases for venting at this value is to:

- [ ] A prevent the Hardened Vent line rupture diaphragm from rupturing.
- [ ] B preserve the structural capability of the Primary Containment hatches.
- [x] C maintain the ability to operate Safety Relief Valves (SRVs).
- [ ] D reduce pressure to the safe side of the Drywell Spray Initiation Limit curve.

Explanation of Answer

A. Incorrect - Rupture Diaphragm setpoint is 30 psig.
B. Incorrect - Bases for PCPL-B.
C. Correct - 60 psig is max DW # that N2 press can operate SRVs.
D. Incorrect - Trips do not identify controlling below this curve as strategy.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.8 SRO Val: 4.0 55.43 ☐

System: 295010 High Drywell Pressure

KA Group Num: AK3 Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.01 Drywell Venting

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip Curves, Tables and Limits	T-Bases	22	19-20	3	
PBAPS Trips	PLOT1560	C	18	8	9

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Question Data for Test: 2001 SRO

Question: 225

A small steam leak has occurred in the drywell on Peach Bottom Unit 3. All drywell temperature indications are available.

Which of the following describes the proper use of available indications to determine actions for this high drywell temperature transient?

- [x] A Bulk average temperature indication is used to determine entry into the TRIP procedure.
- [ ] B Bulk average temperature indication is used to determine RPV Level Instrument availability.
- [ ] C Highest indicated temperature point is used to determine entry into the TRIP procedure.
- [ ] D Highest indicated temperature point is used to determine Drywell Spray Limit Curve conditions.

Explanation of Answer

A. Correct - Bulk average indication or calculated (RT) used for ON and TRIP entry.
B. Incorrect - Points 126/127 used for instrument availability
C. Incorrect - If bulk average and manual calculated average is not available then pt. 136 (Reactor Coolant Pump Area Point) is used.
D. Incorrect - Bulk average is used for saturation curve/Pts 126/127 used for Drywell Spray intiation

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.8 SRO Val: 3.9 55.43 ☑

System:	295012	High Drywell Temperature
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.01	Drywell Temperature

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
T-102 Primary Ctmt Control Bases	T-102 Bases		1	15	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Drywell Temp. Monitoring	RT-O-40C-530			3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIP Procedures	PLOT1560	C	18	8	9

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Question Data for Test: 2001 SRO

Question: 226	Peach Bottom Unit 3 is operating at full power when a Safety Relief Valve fails full open and cannot be reclosed. Torus temperature is 82 degrees F and rising. Continued torus temperature rise may:
☐ A	be prevented by placing one loop of Torus Cooling in service.
☐ B	be prevented by placing both loops of Torus Cooling in service.
☐ C	NOT be prevented unless power is reduced below 25% regardless of Torus Cooling alignment.
☑ D	NOT be prevented while the plant is at power regardless of Torus Cooling alignment.
Explanation of Answer	A. Incorrect - SRV heat input exceeds Torus Cooling Capacity. B. Incorrect - SRV heat input exceeds Torus Cooling Capacity. C. Incorrect - Reducing power will only slightly lower pressure and is accomplished to try to shut the SRV. It does not greatly reduce the heat input of an open SRV. D. Correct - The plant must be shutdown and depressurzied to prevent the torus from continuing to heat up.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.7 55.43 ☐

System:	295013	High Suppression Pool Temperature
KA Group Num:	AK2	Knowledge of the interrelations between- and the following:
KA Detail Num:	AK2.01	Suppression Pool Cooling

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Inadvertent Opening of a SRV - Ba	OT-114 Bases	2.0/3.0	1/2	9	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010			1	1.b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Proc.	PLOT1540	B	6	6	4

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Question Data for Test: 2001 SRO

Question: 227

Peach Bottom Unit 2 was operating at full power when it experieced a loss of feedwater heating. OT-104 "Positive Reactivity Addition" directs total core flow to be reduced. This flow reduction is discontinued when 60 Mlbm/hr flow is reached to PREVENT:

- ☐ A. FLLLP alarms and potential thermal limit violations.
- ☐ B. deep rod insertion from making the power distribution shift more severe.
- ☐ C. exceeding the limits of the "Feedwater Temperature vs. Power" curve in OT-104.
- ☑ D. entering Region I or II of the Power to Flow Map.

Explanation of Answer

A. Incorrect - Note states that these alarms may be received during flow reduction.
B. Incorrect - This is the reason for reducing flow not the reason for stopping at 60 Mlbm/hr
C. Incorrect - Stopping the flow reduction will not prevent exceeding this curve.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE   RO Grp: 1   SRO Grp: 1   RO Val: 4.0   SRO Val: 4.1   55.43 ☐

System: 295014 — Inadvertent Reactivity Addition

KA Group Num: AA1 — Ability to operate and/or monitor the following as they apply to:

KA Detail Num: AA1.07 — Cold Water Injection

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Positive Reactivity Addition Bases	OT-104 Bases	3.5.2	3	17	
Operational Transient Procedures	PLOT1540	11.B.1	6	6	3

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Question Data for Test: 2001 SRO

Question: 228

A hydraulic ATWS has occurred on Peach Bottom Unit 3. The Control Room Supervisor has directed that control rods be inserted using T-220, "Driving Control Rods During a Failure to Scram".

In accordance with T-220, rods MUST be inserted:

- ☐ A — in any sequence using the Rod Control Switch.
- ☐ B — in the GP-3 shutdown sequence using the Rod Control Switch.
- ☑ C — in any sequence using the Emergency In/Notch Override Switch.
- ☐ D — in the GP-3 shutdown sequence using the Emergency In/Notch Override Switch.

Explanation of Answer

A. Incorrect - T-220 directs using the Emergency In/Notch Override Switch.
B. Incorrect - T-220 directs using the Emergency In/Notch Override Switch.
C. Correct
D. Incorrect - T-220 states that rods may be inserted in any sequence.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE   RO Grp: 1   SRO Grp: 1   RO Val: 3.6   SRO Val: 3.7   55.43 ☐

System: 295015 — Imcomplete Scram

KA Group Num: AK2 — Knowledge of the interrelations between - and the following:

KA Detail Num: AK2.02 — RMCS

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Driving Rods During Failure to Scr	T-220-2	1	1	3	
PBAPS Trips	PLOT1560	C	17	8	7

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Question Data for Test: 2001 SRO

Question: 229

The following Peach Bottom Unit 3 conditions exist following a plant transient and scram initiation:

- Rx power is 2 E-2% on WRNMs.
- Rx pressure is 800 psig.
- Rx water level is -90 inches.
- 28 control rods remained at 48.

Under these conditions the reactor is ________(1)________ and an ATWS ________(2)________.

- [x] A shutdown (1), does exist (2)
- [ ] B shutdown (1), does NOT exist (2)
- [ ] C NOT shutdown (1), does exist (2)
- [ ] D NOT shutdown (1), does NOT exist (2)

Explanation of Answer:
A. Correct - Power is below 1E0% however all rods are not inserted to 04 or below.
B. Incorrect - Rx will not remain shutdown by rods alone.
C. Incorrect - Power is below 1E0% on WRNM.
D. Incorrect - Power is below 1E0% on WRNM & Rx will not remain shutdown by rods alone.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.7 55.43 ☐

System:	295015	Incomplete Scram
KA Group Num:	AA1	Ability to operate and/or monitor the following as they apply to:
KA Detail Num:	AA1.07	Neutron Monitoring System

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control	T-101	Notes	1	17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	B	9	8	8

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Question Data for Test: 2001 SRO

Question: 230

Peach Bottom Units 2 and 3 were operating at full power when a fire was identified. Response to the fire required the execution of SE-10, "Plant Shutdown From the Alternative Shutdown Panels".

Which of the following describes the minimum Technical Specification staffing which will be available to you as the Control Room Supervisor (CRS) to perform this shutdown and control the plant from the Alternative Shutdown Panels.

- [x] A 3 Reactor Operators, 5 Equipment Operators
- [ ] B 4 Reactor Operators, 5 Equipment Operators
- [ ] C 3 Reactor Operators, 6 Equipment Operators
- [ ] D 4 Reactor Operators, 6 Equipment Operators

Explanation of Answer

A. Correct answer per Tech Specs and 10CFR50.54.
B. Fourth RO position is not required.
C. Tech Spec minimum is 5 EOs.
D. Fourth RO is an additional position and not required. There are six posted EO positions per the NOM, but only 5 required per Tech Spec.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	

KA Information

Tier: E/APE   RO Grp: 2   SRO Grp: 1   RO Val: 2.3   SRO Val: 3.4   55.43 ☑

System:	295016	Control Room Abandonment
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.4	Knowledge of Shift Staffing Requirements

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Special Events	PLOT1555	II.B.I.I	7	5	5

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Technical Specifications		5.2.2	5.0-3		

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Plant Shutdown From the Alternativ	SE-10	General	1	12	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual	NOM-C-1	Exh. 1.1:1	1	0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Code of Federal Regulations	10CFR50.54	50.54(m)(2)(i)	48		

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Question Data for Test: 2001 SRO

Question: 231

Peach Bottom Unit 3 was operating at 100% power when the "Main Steam Line Hi Radiation" alarm (318 D-2) was received. OT-103, "Main Steam Line High Radiation" was entered.

The OT-103 Immediate Operator Actions direct a GP-9 power reduction to:

- [x] A. limit Main Stack release rates to acceptable values.
- [ ] B. limit Vent Stack release rates to acceptable values.
- [ ] C. reduce the injection rate of the Hydrogen Water Chemistry System.
- [ ] D. reduce reactor coolant system conductivity to acceptable values.

Explanation of Answer

A. Correct - OT-103 Bases reduction to below alarm value should not result in unacceptable release rates.
B. Incorrect - No direct release to vent stack
C. Incorrect - Bases for directing hydrogen water chemistry trip if malfunctioning.
D. Incorrect - Coolant conductivity levels would not be reduced by lowering power.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE   RO Grp: 2   SRO Grp: 1   RO Val: 3.6   SRO Val: 3.8   55.43 ☐

System: 295017 — High Off-site Release Rate

KA Group Num: AK3 — Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.04 — Power Reduction

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
MSL High Radiation - Bases	OT-103	2.1	1	6	

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Question Data for Test: 2001 SRO

Question: 232

During an accident condition at Peach Bottom, the PRO notices that the Unit 2 Vent Stack radiation is high. Which one of the following identifies a possible source of this release?

- [ ] A Recombiner Building Ventilation Exhaust
- [ ] B Standby Gas Treatment Exhaust
- [ ] C PEARL Building Ventilation Exhaust
- [x] D Radwaste Building Ventilation Exhaust

Explanation of Answer

A. Incorrect - Exhausts to Unit 3 Vent Stack.
B. Incorrect - Exhausts to Main Stack.
C. Inocrrect - Exhausts to Unit 3 Vent Stack.
D. Correct - Exhausts to Unit 2 Vent Stack.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 1 RO Val: 3.6 SRO Val: 4.3 55.43 ☑

System:	295017	High Offsite Release Rate
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.04	Source of Offsite Release

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Vent Stack High Rad	ON-104	2.1.1	1	9	
P&ID Prim & Secon.Iso. Control	M-391		2	30	

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Question Data for Test: 2001 SRO

Question: 233

Peach Bottom Unit 3 was operating at full power when the "A" Turbine Building Closed Cooling Water (TBCCW) pump tripped on an electrical fault in the motor. The "B" TBCCW pump is blocked.

Determine the impact on continued power operations.

- ☐ A — A reactor power reduction will be required due to a loss of Main Generator Hydrogen Cooling.
- ☑ B — A reactor power reduction will be required due to a loss of Isophase Bus Cooling.
- ☐ C — An immediate plant shutdown will be required due to a loss of cooling to the CRD Pumps.
- ☐ D — An immediate plant shutdown will be required due to a loss of Instrument Air to the Outboard MSIVs.

Explanation of Answer:

A. Incorrect - Hydrogen coolers are supplied by service water.
B. Correct
C. Incorrect - The CRD pumps receive cooling from RBCCW on a loss of TBCCW.
D. Incorrect - The Air Compressors receive cooling from RBCCW on a loss of TBCCW.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.1 SRO Val: 3.2 55.43 ☐

System: 295018 Partial or complete loss of Component Cooling Water

KA Group Num: AK3 — Knowledge of the reasons for the following responses as they apply to partial or complete loss of Component Cooling Water.

KA Detail Num: AK3.07 — Cross-connecting with backup systems.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Turbine Building Closed Co	ON-118 Bases			4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Turbine building Closed Cooling W	PLOT5034	VI.B.2	12	0	3

Question Data for Test: 2001 SRO

Question: 234

A reactor startup is in progress on Peach Bottom Unit 3. The following conditions exist:

- One condensate pump is in service.
- Three condensate demineralizers are in service.
- The condensate system is lined up for "Long Path Recirc for Startup Level Control"
- CV-2110, "Condensate Recirc Flow" valve is partially open and controlling in automatic.
- The instrument air line to the "Condensate Recirc Flow" valve CV-2110 breaks.

Which of the following describes the impact of this failure?

The CV-2110 "Condensate Recirc Flow" valve fails:

- ☐ A — full closed, causing reactor level to rise.
- ☐ B — full open, causing reactor level to lower.
- ☑ C — full closed, causing the condensate pump to overheat.
- ☐ D — full open, causing condensate pump motor damage due to high current.

Explanation of Answer:

This is similar to an actual Peach Bottom Event.
A. Incorrect - level will be controlled automatically by the start up level controller
B. Incorrect - CV-2110 fails closed on a loss of air.
C. Correct
D. Incorrect - CV-2110 fails closed on a loss of air.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE — RO Grp: 2 — SRO Grp: 2 — RO Val: 3.2 — SRO Val: 3.2 — 55.43 ☐

System:	295019	Partial or complete loss of instrument air.
KA Group Num:	AK2	Knowledge of the interrelations between - and the following:
KA Detail Num:	AK2.07	Condensate System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met	

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensate	PLOT5005	II.E.2.a.1	23	0	6.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Instrument Air	ON-119	Attachment 1	42	14	

Question Data for Test: 2001 SRO

Question: 235

A loss of instrument air transient occurs on Peach Bottom Unit 3. In accordance with ON-119 "Loss of Instrument Air", the reactor must be scrammed if any rod begins to drift in due to lowering scram pilot air header pressure.

What is the bases for this direction?

- [ ] A. To ensure that the scram discharge volume is fully isolated during the scram.
- [x] B. To ensure that various scram valve opening pressures do not result in a random rod pattern.
- [ ] C. To ensure that the individual control rod scram inlet valves do not open before the scram outlet valves.
- [ ] D. To ensure that sufficient volume exists in the scram discharge volume to complete a full scram.

Explanation of Answer

A. Incorrect - Loss of air will result in SDV isolation.
B. Correct - To avoid random rod insertion due to varying scram valve opening pressures.
C. Incorrect - Scram outlet valves open prior to scram inlet valves due to greater spring preload.
D. Incorrect - An automatic scram would be initiated off SDV high level PRIOR to there being insufficient volume in the SDV.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.4 SRO Val: 3.6 55.43 ☑

System:	295019	Partial or Complete Loss of Instrument Air
KA Group Num:	2.4	Emergency Procedure/Plan
KA Detail Num:	2.4.11	Knowledge of abnormal condition procedures.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam Question Source:

Ques Mod Met: Minor wording enhancement.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Instrument Air - Bases	ON-119	Step 2.1 Bases	2		

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550			7	3

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Question Data for Test: 2001 SRO

Question: 236

Unit 2 has experienced a loss of shutdown cooling. ON-125 "Loss of Shutdown Cooling" directs you to determine the expected decay heat load using Operator Aid 95-04 located on the back of Panel 20C005A.

The information necessary to determine expected heat load using this Operator Aid is:

- ☐ A current heat up rate.
- ☐ B current WRNM indicated power.
- ☐ C power history before shutdown.
- ☑ D elapsed time since shutdown.

Explanation of Answer: The operator aid relates time since shutdown to decay heat load in megawatts.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 3 SRO Grp: 2 RO Val: 3.6 SRO Val: 3.8 55.43 ☐

System: 295021 — Loss of Shutdown Cooling

KA Group Num: AK1 — Knowledge of the operational implications of the following concepts as they relate to a loss of shutdown cooling.

KA Detail Num: AK1.01 — Decay Heat

Question Source Information

Ques Source: 1999 PBAPS NRC Exam Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Shutdown Cooling - Bases	ON-125	Step 2.8.7	11	1	
Expected Decay Heat Operator Aid	OP Aid 95-04				

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550			7	3

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Question Data for Test: 2001 SRO

Question: 238

Peach Bottom Unit 2 has experienced a refueling accident. Fission products were released from a dropped and damaged fuel assembly.

The automatic isolation/initiation of Reactor Building Ventilation and Standby Gas Treatment are expected due to:

- [ ] A. Equipment Cell Exhaust High Radiation.
- [ ] B. Refueling Floor Area High Radiation.
- [x] C. Refueling Floor Vent Exhaust High Radiation.
- [ ] D. Fuel Storage Pool High Radiation.

Explanation of Answer

A. Incorrect - Equipment Cell Exhaust High Radiation is not an isolation/initiation condition.
B. Incorrect - Refuel Floor ARM does not cause isolation/initiation
C. Correct - Vent Rad High is an isolation/initiation signal and could occur with release of fission products.
D. Incorrect - Fuel Pool ARM does not cause isolation/initiation

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE   RO Grp: 3   SRO Grp: 1   RO Val: 3.3   SRO Val: 3.6   55.43 ☐

System: 295023 — Refueling Accidents

KA Group Num: AK3 — Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.03 — Ventilation Isolation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Fuel Flr and Fuel Handling Prob.	ON-124 Bases	2.4.5	5	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550	B	6	7	2

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550	B	6	7	3

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 218	218 A-1				

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Question Data for Test: 2001 SRO

Question: 239

A Refuel Outage is in progress on Unit 2. The initial RPV Cavity Floodup is in progress per GP-6, "Refueling Operations". The Fuel Pool Gates are installed and RPV level is 420" and being raised.

The following series of events occur:
- Fuel Pool level starts to lower.
- Skimmer Surge Tank Low Level Alarm is received.
- Refuel Floor Area Radiation Monitors are reading 500 mr/hr (Initial reading was 2 mr/hr).
- Fuel Pool Level stabilizes at 232 ft. 9 inches when level reaches the Skimmer Surge Tank weir.
- Unit 2 Vent Stack Radiation reading on RI-2979A/B are 2 E-4 uci/cc. (The Hi Hi Alarm setpoint value is 5 E-5 uci/cc.)
- Offsite dose calculation rates for TPARD is .074 mrem/hr and .152 mrem/hr for child thyroid CDE.

Use the attached ERP-101"Classification of Emergencies" to determine the classification, if any, for these conditions?

- [x] A Unusual Event
- [ ] B Alert
- [ ] C Site Area Emergency
- [ ] D General Emergency

Explanation of Answer:
A. Correct, EAL 1.2.1.a and 1.2.1.b.
B. Incorrect - Conditions not met for rad level or pool levels, fuel is still covered. <458" but Fuel Pool Gates are installed and Rx Cavity has not been flooded at this time.
C. Incorrect - below the EAL thresholds per EAL 5.1.
D. Incorrect - below the EAL threshold

Exam Level	Cognitive Level	Facility	Materials
SRO	Application	PBAPS	ERP-101 Classification of Emergencies

KA Information

Tier: E/APE RO Grp: 3 SRO Grp: 1 RO Val: 3.2 SRO Val: 4.6 55.43 ☑

System:	295023	Refueling Accidents
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.05	Entry Conditions of Emergency Plan

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Director Training	PEPP6010	II.B	4	3	1.b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Classification of Emergencies	ERP-101	Attachment 1	7	22	

Effective Date:______________

PECO NUCLEAR
PEACH BOTTOM UNITS 2 AND 3
EMERGENCY RESPONSE PROCEDURE

(This is a complete rewrite)

ERP-101 CLASSIFICATION OF EMERGENCIES

1.0 RESPONSIBILITIES

1.1 Shift Management:

1.1.1 Recognize and classify an event or condition.

1.1.2 Assume duties of Emergency Director (ED).

1.2 Plant Manager or designated alternate:

1.2.1 Relieve acting ED.

1.2.2 Assume duties of ED.

2.0 INITIAL ACTIONS

NOTE
THE JUDGMENT OF THE EMERGENCY DIRECTOR TAKES PRECEDENCE OVER GUIDANCE IN THE PROCEDURE.

NOTE
IDENTIFICATION AND CLASSIFICATION OF EMERGENCIES SHOULD BE ACCOMPLISHED WITHIN 15 MINUTES AFTER THE APPLICABLE EMERGENCY ACTION LEVELS (EALs) ARE MET.

2.1 Emergency Director shall:

2.1.1 Select categories appropriate for station events or conditions.

2.1.2 Review Emergency Action Level (EALs) for categories selected.

2.1.3 IF the event trigger is known to be spurious, THEN do not classify the event (i.e., false high reading, false radiation monitor readings, etc.)

2.1.4 Classify the event based on selected categories and most severe EALs.

2.1.5 IF the event or condition classifies as an emergency, THEN assume duties of ED and implement ERP-200.

3.0 CONTINUING ACTIONS

> **NOTE**
> IT IS PREFERABLE TO OBTAIN EMERGENCY RESPONSE MANAGER (ERM) CONCURRENCE PRIOR TO DE-ESCALATION.

3.1 IF emergency conditions dictate,
THEN escalate or de-escalate emergency classification.

4.0 FINAL CONDITIONS

4.1 Emergency conditions have been terminated, or ERP-C-1900, Recovery Phase Implementation has been implemented.

5.0 ATTACHMENTS AND APPENDICES

5.1 Attachment 1 - EAL Table of Contents and Tables 1 through 9. **CM-1, CM-2, CM-3, CM-5**

5.2 Attachment 2 - Terms and Definitions

6.0 SUPPORTING INFORMATION

6.1 Purpose

6.1.1 To provide the method for classifying an event or condition into one of four (4) emergency classifications described in the Nuclear Emergency Plan.

6.1.2 To provide pre-determined Protective Action Recommendations (PARs) for specific plant conditions whenever a General Emergency is declared.

6.2 Criteria For Use

6.2.1 Implement whenever conditions meet or exceed EALs listed in the Tables.

> **NOTE**
> ISSUANCE OF A PAR REQUIRES A GENERAL EMERGENCY CLASSIFICATION AND CONVERSELY A GENERAL EMERGENCY CLASSIFICATION REQUIRES THE ISSUANCE OF A PAR.

6.2.2 PAR information in the tables, is expected to be used when an event rapidly progresses to a General Emergency or when the PAR is based only on plant conditions. Dose Assessment based PAR information may be obtained from the Dose Assessment Coordinator or the Dose Assessment Team Leader. In either case, the most conservative PAR available is to be used.

6.2.3 Whenever the Emergency Operations Facility (EOF) is activated, then all PAR information from the ED should be submitted to the ERM. **CM-4**

6.3 Special Equipment

None

6.4 References

6.4.1 EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

6.4.2 ERP-200, Emergency Director (ED)

6.4.3 ERP-C-1900, Recovery Phase Implementation

6.4.4 Nuclear Emergency Plan

6.4.5 NUMARC/NESP-007, Methodology for Development of Emergency Action Levels

6.4.6 NUREG 0654, FEMA-REP-1, Criteria for Preparations and Evaluation of Radiological Emergency Response Plans in Support of Nuclear Power Plants

6.4.7 PBAPS Technical Specifications

6.4.8 PBAPS Offsite Dose Calculation Manual

6.4.9 PBAPS Updated Final Safety Analysis Report

6.4.10 Reference Manual: Identification and Evaluation of Potentially Reportable Items

6.4.11 SE-1, Plant Shutdown from the Remote Shutdown Panel

6.4.12 SE-5, Earthquake

6.4.13 SE-10, Plant Shutdown from the Alternative Shutdown Panels

6.4.14 T-101, Reactor Pressure Vessel Control

6.4.15 T-102, Primary Containment Control

6.4.16 T-103, Secondary Containment Control

6.4.17 T-104, Radioactivity Release Control

6.4.18 T-116, RPV Flooding

6.4.19 T-200, Primary Containment Venting

6.4.20 SO 67.7A, Verification of Suspected Earthquake or Seismic System Activation

6.4.21 US NRC Regulatory Guide 1.101, Emergency Planning and Preparedness for Nuclear Power Reactors

6.4.22 US NRC Response Technical Manual

6.5 <u>Commitment Annotation</u>

6.5.1 CM-1, NRC Inspection Report 50-277, 278/ 88-12/12 (T00349), (see Attachment 1, tables 1 through 9)

6.5.2 CM-2, Event INV Report 3-90-031, corrective action #7, (T00826), (see Attachment 1, table 1 for Reactor Fuel and table 3 for Fission Product Barrier)

6.5.3 CM-3, NRC URI 85-17-03, IN Inspection Report 86-06/06, (T01934), (see Attachment 1, table 9)

6.5.4 CM-4, Peach Bottom Inspection Report 92-19/19 (T02540), (see section 6.2.3)

6.5.5 CM-5, NRC Inspection 92-03/03, (T02541), (see Attachment 1, table 3 for Fission Product Barrier)

Attachment 1
EAL Table of Contents

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- 1.1 Coolant Activity 6
- 1.2 Irradiated Fuel or New Fuel 7

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- 2.1 Reactor Water Level 8
- 2.2 Reactor Power 9

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MODE	
1	Run
2	Startup
3	Shutdown (hot)
4	Shutdown (cold)
5	Refueling
D	Defueled

CM-1, All Tables

1.0 Reactor Fuel

1.1 Coolant Activity

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Fuel Clad Degradation **1.1.1.a** Applicable Modes: ALL Reactor Coolant activity **> *4 μCi/gm*** Dose Equivalent Iodine 131 **1.1.1.b** Applicable Modes: 1, 2, 3 SJAE Discharge Radiation **> *$2.5x10^3$ mR/hr***
ALERT	*None*
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Unexpected Rise in Plant Radiation or Airbome Concentration. **1.2.1.a** Applicable Modes: ALL Uncontrolled water level drop in the spent fuel pool with all irradiated fuel assemblies remaining covered by water **1.2.1.b** Applicable Modes: ALL Unexpected Skimmer Surge Tank low level alarm **AND** Visual observation of an uncontrolled water level drop below the fuel pool skimmer surge tank inlet IC Unexpected Rise in Plant Radiation **1.2.1.c** Applicable Modes: ALL Radiological readings exceed **600 mR/hr** one foot away **OR** **1200 mR/hr** at the external surface of any dry storage system
ALERT	IC Major Damage to Irradiated Fuel, or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel **1.2.2.a** Applicable Modes: ALL Unplanned general area radiation ***> 500 mR/hr*** on the refuel floor (Table 1-1) **1.2.2.b** Applicable Modes: ALL Report of visual observation of irradiated fuel uncovered **1.2.2.c** Applicable Modes: 5 (With Reactor Refueling Cavity Flooded) Water Level ***< 458" above RPV instrument zero*** for the Reactor Refueling Cavity that will result in Irradiated Fuel uncovering **1.2.2.d** Applicable Modes: ALL Water Level ***< 232ft 3 inches plant elevation*** for the Spent Fuel Pool that will result in Irradiated Fuel uncovering
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

Table 1-1 Refuel Floor ARMs

3-7 (7-9)	Steam Separator Pool
3-8 (7-10)	Refuel Slot
3-9(7-11)	Fuel Pool
3-10(7-12)	Refueling Bridge

2.0 Reactor Pressure Vessel

2.1 Reactor Water Level

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Reactor Coolant System Leakage **2.1.1** Applicable Modes: 1, 2, 3, 4 The following conditions exist: Unidentified Primary System Leakage **> *10 gpm*** into the Drywell **OR** Identified Primary System Leakage **> *25 gpm*** into the Drywell
ALERT	*None*
SITE AREA EMERGENCY	IC Loss of Water Level in the Reactor Vessel That Has or Will Uncover fuel in the Reactor Vessel **2.1.3** Applicable Modes: 4, 5 RPV level **< *-172 "***
GENERAL EMERGENCY	*None*

2.0 Reactor Pressure Vessel

2.2 Reactor Power

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	*None*
ALERT	IC Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was Successful **2.2.2** Applicable Modes: 1, 2 Automatic RPS SCRAM should occur due to RPS Setpoint being exceeded **AND** Failure of Automatic RPS SCRAM to make Reactor shutdown
SITE AREA EMERGENCY	IC Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful **2.2.3** Applicable Modes: 1, 2 RPS SCRAM should occur due to RPS Setpoint being exceeded **AND** Failure of Automatic RPS, ARI **AND** Manual SCRAM to reduce reactor power **< 4%**
GENERAL EMERGENCY	IC Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core **2.2.4** Applicable Modes: 1, 2 RPS SCRAM should occur due to RPS Setpoint being exceeded **AND** Failure of Automatic RPS, ARI **AND** Manual SCRAM to reduce reactor power **< 4%** **AND** Torus Temperature is on the "UNSAFE" side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1) **OR** RPV level **<-200 "** *****PAR***** Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles.

3.0 Fission Product Barrier Table

3.1 Initiating Condition Matrix

USE TABLE 3.2, "FISSION PRODUCT BARRIER STATUS TABLE" FOR CLASSIFYING EVENT

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	**3.1.1** Applicable Modes: 1, 2, 3 ANY Loss **OR** ANY Potential Loss of Primary Containment
ALERT	**3.1.2** Applicable Modes: 1, 2, 3 ANY Loss **OR** ANY Potential Loss of EITHER Fuel Clad **OR** RCS
SITE AREA EMERGENCY	**3.1.3** Applicable Modes: 1, 2, 3 Loss of BOTH Fuel Clad **AND** RCS **OR** Potential Loss of BOTH Fuel Clad **AND** RCS **OR** Potential Loss of EITHER Fuel Clad **OR** RCS, **AND** Loss of ANY Additional Barrier
GENERAL EMERGENCY	**3.1.4** Applicable Modes: 1, 2, 3 Loss of ANY Two Barriers **AND** Potential Loss of Third Barrier ***PAR*** Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles. (See Fission Product Barrier Table 3.2 for exception based on extremely Hi Containment Radiation Levels.)

NOTES:

1. If a "Loss" condition is satisfied, the "Potential Loss" category can be considered satisfied. This is accounted for in the matrix contained in the Fission Product Barrier Table 3.2 used to determine the proper classification based on Fission Product Barrier status.

2. For all conditions listed in Fission Product Barrier Table 3.2, the barrier failure column is only satisfied if it fails when called upon to mitigate an accident. For example, failure of both containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal power operations, it will be an active Technical Specification Action Statement. However, during accident conditions, this will represent a breach of containment.

3.2 Fission Product Barrier Status Table
Applicable Modes: 1, 2, 3

Barrier / Parameter	Fuel Clad Loss	Fuel Clad Potential Loss	Reactor Coolant System Loss	Reactor Coolant System Potential Loss	Primary Containment Loss	Primary Containment Potential Loss
Reactor Coolant Activity	Reactor Coolant activity > ***300 μCi/gm*** Dose Equivalent Iodine 131	N/A	N/A	N/A	N/A	N/A
RPV Level	RPV level < ***-200 "***	RPV level < ***-172 "***	RPV level < ***-172 "***	N/A	N/A	RPV level cannot be restored above ***-200 "*** within the time limit of the "SAFE" region of the Maximum Core Uncovery Time Limit Curve (T-116, RF-1)
RPV Level Unknown	N/A	N/A	N/A	RPV level cannot be determined	N/A	RPV level cannot be determined **AND** RPV Flooding cannot be established as indicated by inability to maintain 5 ADS/SRVs open with RPV pressure at least 60 psig above Torus pressure per T-116
RCS Leak Rate	N/A	N/A	N/A	RCS leakage ***>50 gpm***	N/A	N/A
Drywell Pressure	N/A	N/A	Drywell Pressure ***> 2.0 psig*** **AND** Indication of a leak inside drywell	N/A	Rapid, unexplained drop in Drywell Pressure following initial rise **OR** Drywell pressure response not consistent with LOCA conditions	Drywell Pressure ***> 49 psig*** and rising **OR** Drywell Hydrogen > ***6%*** **AND** Drywell Oxygen > ***5%***
Drywell Radiation	Drywell Rad Monitor reading ***> 8x10⁴ R/hr***	N/A	Drywell Rad Monitor reading ***> 15 R/hr***	N/A	N/A	Drywell Rad Monitor reading ***> 6x10⁵ R/hr*** ***PAR*** **Evacuate 5 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 5-10 miles.**

3.2 Fission Product Barrier Status Table
Applicable Modes: 1, 2, 3

Barrier / Parameter	Fuel Clad		Reactor Coolant System		Primary Containment	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
Containment Isolation	N/A	N/A	N/A	Unisolable primary system leakage outside drywell as indicated by T-103, **Temperature Action Level** is exceeded in ONE area requiring a SCRAM **<u>OR</u>** Unisolable primary system leakage outside drywell as indicated by T-103, **Radiation Action Level** is exceeded in ONE area requiring a SCRAM	Failure of both valves in any one line to close **<u>AND</u>** downstream pathway to the environment exists **<u>OR</u>** Intentional venting per T-200 is required **<u>OR</u>** Unisolable primary system leakage outside drywell as indicated by T-103, **Temperature Action Level** is exceeded in ONE area requiring a SCRAM **<u>OR</u>** Unisolable primary system leakage outside drywell as indicated by a T-103, **Radiation Action Level** is exceeded in ONE area requiring a SCRAM	N/A
Emergency Director Judgment	Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the FUEL CLAD barrier		Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the RCS barrier		Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the Primary Containment barrier	

In the table below, circle all of the appropriate X's in each applicable row for each Loss or Potential Loss of Fission Product Barrier as determined by the table above.

Classify the event as identified in the table heading if all X's in a column under that heading are circled.

Fission Product Barrier Status	**Unusual Event**		**ALERT**				**SITE AREA EMERGENCY**								**GENERAL EMERGENCY**			
Fuel Clad - Loss			X				X		X		X				X	X		X
Fuel Clad - Potential Loss				X				X		X		X					X	
Reactor Coolant System - Loss					X		X			X			X		X	X	X	
Reactor Coolant System-Potential Loss						X		X	X					X				X
Primary Containment - Loss	X										X	X	X	X	X		X	X
Primary Containment - Potential Loss		X														X		

******PAR******
Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles. (Upgrade PAR for D/W Rad > *6x10⁵ R/hr*)

4.0 Secondary Containment Bypass

4.1 Main Steam Line

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Fuel Clad Degradation **4.1.1** Applicable Modes: 1, 2, 3 Main Steam Line HiHi Radiation (10xNFPB)
ALERT	IC RCS Leak Rate **4.1.2** Applicable Modes: 1, 2, 3 Indication of a Main Steam Line Break: Hi Steam Flow Annunciator **AND** Hi Steam Tunnel Temperature Annunciator **OR** Direct report of steam release
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

5.0 Radioactivity Release

5.1 Effluent Release and Dose

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Technical Specifications for 60 Minutes or Longer **5.1.1.a** Applicable Modes: ALL A valid reading on one or more of the following radiation monitors that exceeds **TWO TIMES** the HiHi alarm setpoint value for ***> 60 minutes:*** Main Stack, Vent Stack, Radwaste Discharge, Service Water Discharge **<u>AND</u>** Calculated maximum offsite dose rate using computer dose model exceeds ***0.114 mRem/hr TPARD <u>OR</u> 0.342 mRem/hr child thyroid CDE*** based on a 60 minute average Note: If the required dose projections cannot be completed within the 60 minute period, then the declaration must be made based on the valid sustained monitor reading. **5.1.1.b** Applicable Modes: ALL Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates exceeding **TWO TIMES** Tech Specs (Liquid Release ODCM 3.8.B.1 and Gaseous Release ODCM 3.8.C.1.b) for ***> 60 minutes***
ALERT	IC Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Radiological Technical Specifications for 15 Minutes or Longer **5.1.2.a** Applicable Modes: ALL A valid reading on one or more of the following radiation monitors that exceeds **TWO HUNDRED TIMES** the HiHi alarm setpoint value for ***> 15 minutes:*** Main Stack, Vent Stack, Radwaste Discharge, Service Water Discharge **<u>AND</u>** Calculated maximum offsite dose rate exceeds **11.4 mRem/hr TPARD <u>OR</u> 34.2 mRem/hr child thyroid CDE** based on a 15 minute average Note: If the required dose projections cannot be completed within the 15 minute period, then the declaration must be made based on the valid sustained monitor reading. **5.1.2.b** Applicable Modes: ALL Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates exceeding **TWO HUNDRED TIMES** Tech Specs (Liquid Release ODCM 3.8.B.1 and Gaseous Release ODCM 3.8.C.1.b) for ***> 15 minutes***

SITE AREA EMERGENCY	**IC** Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR Whole Body or 500 mR Child Thyroid for the Actual or Projected Duration of the Release **5.1.3** Applicable Modes: ALL A valid reading on one or more of the following radiation monitors that exceeds or is expected to exceed the value shown for ***> 15 minutes*** **<u>AND</u>** Dose Projections are not available: Main Stack 5.84 µCi/cc Vent Stack 2.08E-3 µCi/cc Torus Vent 203 cpm Note: If the required dose projections cannot be completed within the 15 minute period, then the declaration must be made based on the valid sustained monitor reading. <u>OR</u> Projected offsite dose using computer dose model exceeds ***100 mRem TPARD*** **<u>OR</u>** ***500 mRem*** **child thyroid CDE** <u>OR</u> Analysis of Field Survey results indicate site boundary whole body dose rate exceeds ***100 mRem/hr*** expected to continue for more than one hour, <u>OR</u> Analysis of Field Survey results indicate child thyroid dose commitment of ***500 mRem*** for one hour of inhalation
GENERAL EMERGENCY	**IC** Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mR Whole Body or 5000 mR Child Thyroid for the Actual or Projected Duration of the Release Using Actual Meteorology **5.1.4** Applicable Modes: ALL A valid reading on one or more of the following radiation monitors that exceeds or is expected to exceed the value shown for ***> 15 minutes*** **<u>AND</u>** Dose Projections are not available: Main Stack 58.4 µCi/cc Vent Stack 2.08E-2 µCi/cc Torus Vent 2000 cpm Note: If the required dose projections cannot be completed within the 15 minute period, then the declaration must be made based on the valid sustained monitor reading. <u>OR</u> Projected offsite dose using computer dose model exceeds ***1000 mRem TPARD*** **<u>OR</u>** ***5000 mRem*** **child thyroid CDE** <u>OR</u> Analysis of Field Survey results indicate site boundary whole body dose rate exceeds ***1000 mRem/hr*** expected to continue for more than one hour, <u>OR</u> Analysis of Field Survey results indicate child thyroid dose commitment of ***5000 mRem*** for one hour of inhalation ***PAR*** Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles.

NOTE: CDE = Committed Dose Equivalent, TPARD = Total Protective Action Recommendation Dose

5.0 Radioactivity Release

5.2 In-Plant Radiation

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Unexpected Rise in Plant Radiation or Airborne Concentration **5.2.1** Applicable Modes: ALL Valid Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels * Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.
ALERT	IC Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown **5.2.2.a** Applicable Modes: ALL Valid radiation level readings ***> 5000 mR/hr*** in areas requiring infrequent access to maintain plant safety functions as identified in procedure SE-1, SE-10 **AND** Access is required for safe plant operation, but is impeded, due to radiation dose rates **5.2.2.b** Applicable Modes: ALL Valid Control Room **OR** Central Alarm Station radiation reading ***> 15 mR/hr***
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

This Intentionally Left Blank

6.0 Loss of Power

6.1 Loss of AC or DC Power

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes **6.1.1.a** Applicable Modes: ALL The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer for ***>15 minutes*** **AND** At least ***Two*** Diesel Generators are supplying power to their respective 4 KV emergency busses IC Unplanned Loss of Required DC Power During Cold Shutdown or Refueling Mode for Greater than 15 Minutes **6.1.1.b** Applicable Modes: 4, 5 Unplanned Loss of ALL safety related DC Power indicated by ***< 107.5 VDC*** on DC Panels 2(3)0D21, 22, 23, 24 for ***>15 minutes***
ALERT	IC AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout **6.1.2.a** Applicable Modes: 1, 2, 3 The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer for ***>15 minutes*** **AND** Only ***One*** 4 KV emergency bus powered from a Single Onsite Power Source due to the Loss of: Three of Four Division Diesel Generators, D/G Output Breakers, or 4 KV Emergency Busses as indicated by bus voltage IC Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses During Cold Shutdown Or Refueling Mode **6.1.2.b** Applicable Modes: 4, 5, D The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer **AND** Failure to restore power to at least ***One*** 4 KV emergency bus ***within 15 minutes*** from the time of loss of both offsite and onsite AC power

SITE AREA EMERGENCY	**IC** Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses **6.1.3.a** Applicable Modes: 1, 2, 3 The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer **AND** Failure to restore power to at least ***One*** 4 KV emergency bus ***within 15 minutes*** from the time of loss of both offsite and onsite AC **IC** Loss of All Vital DC Power **6.1.3.b** Applicable Modes: 1, 2, 3 Loss of ALL Safety Related DC Power indicated by ***< 107.5 VDC*** on DC Panels 2(3)0D21, 22, 23, 24 for ***> 15 minutes***
GENERAL EMERGENCY	**IC** Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power **6.1.4** Applicable Modes: 1, 2, 3 Prolonged loss of all offsite and onsite AC power as indicated by: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer **AND** Failure of ALL Emergency Diesel Generators to supply power to 4 KV emergency busses **AND** At least one of the following conditions exist: • Restoration of at least ***One*** emergency bus within ***2 hours*** is ***NOT*** likely **OR** • Reactor Water Level cannot be maintained ***> -172 "*** **OR** • Torus temperature is on the **"UNSAFE"** side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1) ****PAR*** Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles.

7.0 Internal Events

7.1 Technical Specification & Control Room Evacuation

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Inability to Reach Required Shutdown Mode Within Technical Specification Limits **7.1.1** Applicable Modes: 1, 2, 3 Inability to reach required shutdown mode within Tech. Spec. LCO required action completion time.
ALERT	IC Control Room Evacuation Has Been Initiated **7.1.2** Applicable Modes: ALL Entry into SE-1 or SE-10 procedure for Control Room evacuation
SITE AREA EMERGENCY	IC Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established **7.1.3** Applicable Modes: ALL The following conditions exist: Control room evacuation has been initiated **AND** Control of the plant cannot be established per SE-1or SE-10 within ***15 minutes***
GENERAL EMERGENCY	*None*

7.0 Internal Events

7.2 Loss of Decay Heat Removal Capability

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	*None*
ALERT	**IC** Inability to Maintain Plant in Cold Shutdown **7.2.2** Applicable Modes: 4, 5 The following conditions exist: Unplanned Loss of <u>ALL</u> Tech Spec required systems available to provide Decay Heat Removal functions <u>**AND**</u> Uncontrolled Temperature rise that either: • Exceeds ***212 °F*** (Excluding a <15 minute rise >212° F with a heat removal function restored) <u>**OR**</u> • Results in temperature rise approaching ***212 °F*** (with <u>NO</u> heat removal function restored)
SITE AREA EMERGENCY	**IC** Complete Loss of Function Needed to Achieve or Maintain Hot Shutdown **7.2.3** Applicable Modes: 1, 2, 3 Loss of TORUS heat sink capabilities as evidenced by T-102 T/T legs directing a T-112 Emergency Blowdown
GENERAL EMERGENCY	*None*

7.0 Internal Events

7.3 Loss of Assessment / Communication Capability

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Unplanned Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes **7.3.1.a** Applicable Modes: 1, 2, 3 Unplanned loss of most or all safety system annunciators (Table 7-1) <u>**OR**</u> indicators (Table 7-2) for **> 15 minutes** requiring increased surveillance to safely operate the unit(s). IC Unplanned Loss of All Onsite or Offsite Communications Capabilities **7.3.1.b** Applicable Modes: ALL Loss of ALL Onsite communications (Table 7-3) affecting the ability to perform routine operations <u>**OR**</u> Loss of ALL Offsite communications (Table 7-3)
ALERT	IC Unplanned Loss of Most or All Safety System Annunciation or Indication In Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable **7.3.2** Applicable Modes: 1, 2, 3 Unplanned loss of most or all safety system annunciators (Table 7-1) <u>**OR**</u> indicators (Table 7-2) for **> 15 minutes** requiring increased surveillance to safely operate the unit(s) <u>**AND EITHER**</u> A significant plant transient is in progress (Table 7-4) <u>**OR**</u> the plant monitoring system (PMS) is unavailable.
SITE AREA EMERGENCY	IC Inability to Monitor a Significant Transient in Progress **7.3.3** Applicable Modes: 1, 2, 3 Loss of safety system annunciators (Table 7-1) <u>AND</u> indicators (Table 7-2) <u>AND</u> PMS <u>AND</u> a significant plant transient is in progress. (Table 7-4)
GENERAL EMERGENCY	*None*

Table 7-1 Safety System Annunciators

ECCS
Containment Isolation
Reactor Trip
Process Radiation Monitoring

Table 7-2 Safety Function Indicators

Reactor Power
Decay Heat Removal
Containment Safety Functions

Table 7-3 Communications

	Onsite	Offsite
Site Phones (GTE System)	X	X
OMNI System	X	X
Plant Public Address	X	
Station Radio	X	
NRC (FTS-2000)		X
PA State Police Radio		X
Load Dispatcher Radio		X
PECO Dial Network		X

Table 7-4 Significant Plant Transients

SCRAM
Recirc Runbacks > 25% thermal power
Sustained power oscillations 25% peak to peak
Stuck open relief valve(s)
ECCS injection

8.0 External Events

8.1 Security Threats

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	**IC** Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant **8.1.1** Applicable Modes: ALL Credible sabotage or bomb threat within the Protected Areas **<u>OR</u>** Credible intrusion and attack threat to the Protected Areas **<u>OR</u>** Attempted intrusion and attack to the Protected Areas **<u>OR</u>** Attempted sabotage discovered within the Protected Areas **<u>OR</u>** Hostage/Extortion situation that threatens normal plant operations
ALERT	**IC** Security Event in a Plant Protected Area **8.1.2** Applicable Modes: ALL Intrusion into plant protected areas by a hostile force **<u>OR</u>** Confirmed bomb, sabotage or sabotage device discovered in the Protected Areas
SITE AREA EMERGENCY	**IC** Security Event in a Plant Vital Area **8.1.3** Applicable Modes: ALL Intrusion into plant Vital area by a hostile force **<u>OR</u>** Confirmed bomb, sabotage or sabotage device discovered in a Vital Area
GENERAL EMERGENCY	**IC** Security Event Resulting in Loss of Ability to Reach and Maintain Cold Shutdown **8.1.4** Applicable Modes: ALL Loss of physical control of the control room due to security event **<u>OR</u>** Loss of physical control of all remote shutdown capability due to security event ***PAR*** Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles.

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8.0 External Events

8.2 Fire / Explosion and Toxic / Flammable Gases

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	**IC** Fire Within Protected Area Boundary Not Extinguished Within 15 Minutes of Detection **8.2.1.a** Applicable Modes: ALL Fire within ON-114 Plant Vital Structures (Table 8-1) which is not extinguished within ***15 minutes*** of control room notification or verification of a control room alarm **IC** Release of Toxic or Flammable Gasses Deemed Detrimental to Safe Operation of the Plant **8.2.1.b** Applicable Modes: ALL Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant **OR** Report by Local, County or State Officials for potential evacuation of site personnel based on offsite event **IC** Natural and Destructive Phenomena Affecting the Protected Area **8.2.1.c** Applicable Modes: ALL Report by plant personnel of an unanticipated explosion within protected area boundary resulting in visible damage to permanent structure or equipment
ALERT	**IC** Fire or Explosion Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown **8.2.2.a** Applicable Modes: ALL The following conditions exist: Fire or explosion which potentially makes inoperable: *Two or More* subsystems of a Safe Shutdown System (Table 8-2) **OR** *Two or More* Safe Shutdown Systems **OR** Plant Vital Structures containing Safe Shutdown Equipment **AND** Safe Shutdown System or Plant Vital Structure is required for the present Operational Mode

ALERT	**IC** Release of Toxic or Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown **8.2.2.b** Applicable Modes: ALL Report or detection of toxic gases within Plant Vital Structures (Table 8-1) in concentrations that will be life threatening to plant personnel **OR** Report or detection of flammable gases within Plant Vital Structures (Table 8-1) in concentrations affecting the safe operation of the plant
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

Table 8-1 Plant Vital Structures

Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

Table 8-2 Safe Shutdown Systems

Diesel Generators	4KV Safeguard Buses	ADS
HPCI	RCIC	RHR (All Modes)
Core Spray	HPSW	ESW
SBGTS	ECW	CAC/CAD
PCIS	Control Room Ventilation	

8.0 External Events

8.3 Man-Made Events

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Destructive Phenomena Affecting the Protected Area **8.3.1.a** Applicable Modes: ALL Vehicle crash within protected area boundary that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant. **8.3.1.b** Applicable Modes: ALL Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.
ALERT	IC Destructive Phenomena Affecting the Plant Vital Area **8.3.2** Applicable Modes: ALL Vehicle crash affecting Plant Vital Structures (Table 8-1) **OR** Turbine failure generated missiles result in any visible structural damage to or penetration of any Plant Vital Structures (Table 8-1)
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

Table 8-1 Plant Vital Structures

Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

8.0 External Events

8.4 Natural Events

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	IC Natural and Destructive Phenomena Affecting the Protected Area **8.4.1.a** Applicable Modes: ALL Earthquake ***>.01 g*** as determined by procedure SO 67.7.A **8.4.1.b** Applicable Modes: ALL Report by plant personnel of tornado striking within protected areas **OR** Wind speeds ***> 75 mph*** as indicated on site Meteorological data for ***> 15 minutes*** **8.4.1.c** Applicable Modes: ALL Assessment by the control room that an event has occurred. (Natural and Destructive Phenomena Affecting the Protected Areas) **8.4.1.d** Applicable Modes: All High River level **> 112'** OR Low River level **< 98.5'**
ALERT	IC Natural and Destructive Phenomena Affecting the Plant Vital Area **8.4.2.a** Applicable Modes: ALL Earthquake ***>.05 g*** (Operating Basis Earthquake OBE) as determined by procedure SO 67.7.A **8.4.2.b** Applicable Modes: ALL Tornado or wind speeds ***> 75 mph*** causing damage to Plant Vital Structures (Table 8-1) **8.4.2.c** Applicable Modes: ALL Report of any visible structural damage to any Plant Vital Structure (Table 8-1) **8.4.2.d** Applicable Modes: All High River level **> 116'** OR Low River level **< 92.5'**
SITE AREA EMERGENCY	*None*
GENERAL EMERGENCY	*None*

able 8-1 Plant Vital Structures

Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

9.0 Other

9.1 General

CLASSIFICATION	EMERGENCY ACTION LEVEL
UNUSUAL EVENT	**IC** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Unusual Event **9.1.1** Applicable Modes: ALL Other conditions exist which in the judgment of the Emergency Director indicate a potential degradation of the level of safety of the plant
ALERT	**IC** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert **9.1.2** Applicable Modes: ALL Other conditions exist which in the Judgment of the Emergency Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted
SITE AREA EMERGENCY	**IC** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency **9.1.3** Applicable Modes: ALL Other conditions exist which in the Judgment of the Emergency Director indicate actual or likely major failures of plant functions needed for protection of the public
GENERAL EMERGENCY	**IC** Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency **9.1.4** Applicable Modes: ALL Other conditions exist which in the Judgment of the Emergency Director indicate: (1) actual or imminent substantial core degradation with potential for loss of containment, or (2) potential for uncontrolled radionuclide releases. These releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary ***PAR*** Evacuate 2 mile radius, evacuate affected sector(s) plus 1 sector on each side of affected sector(s) for 2-5 miles.

Attachment 2
TERMS AND DEFINITIONS

EMERGENCY ACTION LEVEL (EAL)	Plant parameters or other condition which if met or exceeded the emergency classification level and requires a declaration of emergency.
OPERABLE	System, subsystem, train, component, or device, and all auxiliaries required for their operation, is capable of performing its specified function in the intended manner.
PROTECTIVE ACTION RECOMMENDATIONS (PAR)	Recommendation made to the state action to be taken to avoid or reduce projected dose to the public.
PROJECTED DOSE	An estimate of radiation dose which affected individuals could potentially receive if protective actions are not taken.
TPARD	Total Protective Action Recommendation Dose. (TPARD = External Dose & Internal Dose & Dose Due to 4-Day Shine)
CDE	Committed Dose Equivalent. (CDE = internal Organ Dose from Ingestion)
CEDE	Committed Effective Dose Equivalent. (CEDE = Internal Whole Body Dose from Ingestion)
TEDE	Total Effective Dose Equivalent. (TEDE = Deep Dose Equivalent & CEDE Dose)
PROTECTIVE ACTION GUIDE (PAG)	Action guidelines based on projections for the total integrated dose a member of the public would receive for the duration of the emergency.
SABOTAGE	An act conducted by a person or persons with the intent of damaging or impairing the operation of the plant.
SECURITY COMPROMISE	A security threat as illustrated by attempted entry or sabotage with the intent to gain physical control of the plant.
UNUSUAL EVENT	Events in progress or have occurred, that indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.
ALERT	Events in progress or have occurred that involve actual or potential substantial degradation of the level of safety of the plant. Any releases of radioactive material are expected to be limited to small fractions of the Environmental Protective Agency (EPA) Protective Action Guidelines (PAG) exposure levels.
SITE AREA EMERGENCY	Events in progress or which have occurred that involve actual or likely major failures of plant functions needed for protection of the public. Any releases of radioactive material are not expected to exceed EPA PAG exposure levels except near site boundary.
GENERAL EMERGENCY	Events in progress or which have occurred that involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases of radioactive material can be reasonably expected to exceed EPA PAG exposure levels off-site for more than the immediate site area.

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Question Data for Test: 2001 SRO

Question: 240

Peach Bottom Unit 3 was operating at 50% power when a small steam leak occurred in the drywell.
-Drywell temperature is 250 degrees F and continuing to rise.
-Drywell pressure is 0.9 psig and steady
-The STA has reported that these conditions indicate a breach in the Primary Containment

The Technical Support Center (TSC) has recommended drywell sprays to reduce containment temperatures. Under these conditions, the Drywell Spray Logic will:

- ☐ A permit spray with the use of a bypass key.
- ☐ B permit spray until a LPCI initiation signal present.
- ☐ C prevent spray due to low torus pressure.
- ☑ D prevent spray due to low drywell pressure.

Explanation of Answer

A. Incorrect - low drywell pressure can not be bypassed
B. Incorrect - drywell spray is not permitted
C. Incorrect - system monitors drywell pressure
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 4.2 SRO Val: 4.2 55.43 ☐

System:	295024	High Drywell Pressure
KA Group Num:	EK2	Knowledge of the interrelations between - and the following:
KA Detail Num:	EK2.11	Drywell Spray (RHR) Logic

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Prints	M-1-S-65	D			

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010			1	4.s

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Question Data for Test: 2001 SRO

Question: 241

Peach Bottom Unit 2 has experienced a Loss of Coolant Accident (LOCA). Preparations are in progress to vent the primary containment due to a high drywell pressure per T-200-2 "Primary Containment Venting". The EO requests direction to either electrically or mechanically position the 6 inch ILRT valves.

Which one of the following identifies the method that may minimize dose to the EO and why dose would be minimized?

- [x] A Electrical positioning is from a lower dose area in Reactor Building.
- [ ] B Electrical positioning is from a remote location outside the Reactor Building.
- [ ] C Mechanical positioning is from a lower dose area in the Reactor Building.
- [ ] D Mechanical positioning is from a remote location outside the Reactor Building.

Explanation of Answer

A. Correct - Electrical positioning is performed at a panel on RB135' East wall
B. Incorrect - Electrical positioning is in RB 135' el.
C. Incorrect - Mechanical is at valve location in the same proximity as the vent line
D. Incorrect - Not possible since the valves are in the reactor building.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.3 SRO Val: 3.5 55.43 ☑

System: 295024 High Drywell Pressure

KA Group Num: 2.4 Emergency Procedures/Plan

KA Detail Num: 2.4.35 Knowledge of local Aux. Operator tasks during emergency operations including system geography and system implications.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Containment Venting via 6" ILRT F	T-200C-2	4	3	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	C	10	8	13

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Question Data for Test: 2001 SRO

Question: 242

Peach Bottom Unit 3 is operating at 100% power when a complete failure of the EHC System occurs. This failure causes an immediate main turbine trip with no bypass valve operation.

In accordance with Chapter 14 of the PBAPS Updated Final Safety Analysis Report (UFSAR), what is the MAXIMUM expected plant response to this transient.

- ☐ A — Five SRVs open.
- ☑ B — All SRVs open.
- ☐ C — All SRVs and both safety valves open, the Safety Limit is NOT violated.
- ☐ D — All SRVs and both safety valves open, the Safety Limit is violated.

Explanation of Answer: The UFSAR specifies that pressure is expected to reach 1249 psig in the bottom head during this transient. This means that steam pressure will be high enough to open all of the SRVs, but not the safety valves. The Safety Limit of 1325 psig steam dome pressure will not be exceeded.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	

KA Information

Tier: E/APE — RO Grp: 1 — SRO Grp: 1 — RO Val: 4.3 — SRO Val: 4.3 — 55.43 ☑

System:	295025	High Reactor Pressure
KA Group Num:	EA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	EA2.01	Reactor Pressure

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control	T-101	Entry		21	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trip Proc.	PLOT1560	C	C	8	2

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS FSAR	UFSAR	14.5.1.2.2	14.5-3	17	

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Question Data for Test: 2001 SRO

Question: 243

Select the following condition in which the Peach Bottom TRIP procedures require a continuous nitrogen supply to be available before Safety Relief Valve (SRV) operation.

- [x] A — To lower pressure to below 1050 psig to stabilize pressure.
- [ ] B — To lower pressure to 950 psig due to SRV cycling.
- [ ] C — To lower pressure to maintain on SAFE side of the Heat Capacity Temperature Limit.
- [ ] D — To perform a T-112 Emergency Blowdown.

Explanation of Answer

A. Correct - This step requires continuous nitrogen supply.
B. Incorrect - Allowed by T-101 Bases.
C. Incorrect - Allowed by T-102 Bases.
D. Incorrect - Per T-112 Bases.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.1 SRO Val: 4.0 55.43 ☑

System:	295025	High Reactor Pressure
KA Group Num:	2.4	Emergency Procedures/Plan
KA Detail Num:	2.4.6	Knowledge of symptom based EOP mitigation strategies.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control	T-101	RC/P-13	25	21	
PBAPS Trip Proc.	PLOT1560	C	17	8	3

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Question Data for Test: 2001 SRO

Question: 244

The bases for entering T-116 "RPV Flooding" procedure during a high drywell temperature condition is to establish:

- [ ] A flooding conditions before SRVs fail due to exceeding the SRV cabling design temperature.
- [x] B adequate core cooling due to indicated level errors from reference leg flashing.
- [ ] C flooding conditions before the level drops below the variable leg tap elevation for the Fuel Zone instruments.
- [ ] D adequate core cooling due to indicated level errors from variable leg flashing.

Explanation of Answer:
A. Incorrect - Bases for 281 degrees F in accordance with T-112.
B. Correct - Ref leg official or saturation temp results in "RPV Cannot be Determined".
C. Incorrect - Minimum indication on one instrument doesn't reqire T-116 unless level cannot be determined,
D. Incorrect - The concern is for reference leg flashing

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.5 SRO Val: 3.8 55.43 ☐

System: 295028 High Drywell Temperature

KA Group Num: EK3 Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: EK3.02 RPV Flooding

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102 Bases	DW/T	19	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trip Proc.	PLOT1560	C	18	8	11

89

Question Data for Test: 2001 SRO

Question: 245

T-102 "Primary Containment Control" provides direction to maintain Torus level in the band of 14.5 ft. to 14.9 ft. In accordance with the TRIP Bases what is the first concern during a rising torus level transient?

- [ ] A Submerging the Reactor Building to Torus Vacuum Breaker Line.
- [x] B Excessive stress on SRV tail pipes.
- [ ] C Submergence of the Torus Spray Header.
- [ ] D Excessive stress on ECCS suction piping.

Explanation of Answer: T/L-18 Basis - Increased submergence of SRV tailpipes can cause excessive stress on SRV pipes, quenchers, and associated supports.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.4 SRO Val: 3.7 55.43

System: 295029 High Suppression Pool Water Level

KA Group Num: EK1 Knowledge of the operational implications of the following as they apply to high suppression pool level.

KA Detail Num: EK1.01 Containment Integrity

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control - Bas	T-102 Bases	T/L-18	8	14	
PBAPS TRIPS	PLOT1560			8	9

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Question Data for Test: 2001 SRO

Question: 246

For a lowering suppression pool level T-102, "Torus Level", directs that if Torus level cannot be maintained above 9.5' secure HPCI. It does not direct that RCIC be secured until < 6'.

What is the basis for securing HPCI but not RCIC at 9.5'?

- [ ] A — HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust becomes uncovered at 6'.
- [x] B — HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust is an insignificant containment input:
- [ ] C — HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust becomes uncovered at 6'.
- [ ] D — HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust is a insignificant containment input.

Explanation of Answer: Self explanatory.
RCIC is secured at 6' if it is aligned to the Torus to prevent vortexing.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE — RO Grp: 2 — SRO Grp: 1 — RO Val: 3.6 — SRO Val: 3.7 — 55.43 ☐

System: 295030 — Low Suppression Pool Water Level

KA Group Num: EK3 — Knowledge of the reasons for the following responses as they apply to low suppression pool water level.

KA Detail Num: EK3.03 — RCIC Operations

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102	/L-11 to T/L-16	8a7-8	14	
PBAPS TRIPS	PLOT1560			8	9

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Question Data for Test: 2001 SRO

Question: 247

Peach Bottom Unit 2 has experienced a Residual Heat Removal System leak lowering Torus level to 11.5 feet. A small steam leak then develops and pressurizes the drywell.

Evaluate these conditions and determine at what approximate drywell pressure you would expect to see a torus pressure rise from this leak? (Assume the drywell and torus were initially at 0.5 psig).

- [ ] A 0.6 psig
- [x] B 1.0 psig
- [ ] C 2.5 psig
- [ ] D 9.5 psig

Explanation of Answer

A. Incorrect - .1 psig is design delta P to operate Vacuum Breakers.
B. Correct - The bottom of the downcomers is at 10.5 feet. With only 1 ft of submergence non-condensables would move to torus when approximately .5 psig differential.
C. Incorrect - 2 psid is delta P with normal level ~14.6-14.9 ft.
D. Incorrect - 9 psi in torus is when 95% of DW N2 is in torus.

Exam Level	Cognitive Level	Facility	Materials
SRO	Application	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 1 RO Val: 3.5 SRO Val: 3.7 55.43 ☑

System:	295030	Low Suppression Pool Water Level
KA Group Num:	EA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	EA2.04	Drywell/Suppression chamber differential pressure: Mark I and II.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control-Base	T-102 Bases	T/L	7	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment	PLOT5007			0	3g

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Question Data for Test: 2001 SRO

Question: 248

Peach Bottom Unit 3 has experienced a scram due to a loss of feedwater transient. With NO injection sources available, level has continued to drop.

For these conditions, what is the LOWEST reactor water level at which adequate core cooling (ACC) is still maintained.

ACC exists ONLY until reactor water level is BELOW:

- [ ] A -172".
- [ ] B -195".
- [x] C -210".
- [ ] D -226".

Explanation of Answer

A. Incorrect - Top of Active Fuel.
B. Incorrect - would be correct if injection was present.
C. Correct - ACC is provided by steam cooling with no RPV injection until -210".
D. Incorrect - This is the level for 2/3 core coverage.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 4.6 SRO Val: 4.7 55.43 ☐

System: 295031 Reactor Low Water Level

KA Group Num: EK1.01 Knowledge of the operational/implications of the following concepts as they apply to reactor low water level.

KA Detail Num: EK1.01 Adequate Core Cooling

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Introduction to TRIPS and SAMPS	T-Bases (Intro)	5.1	18	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIPS	PLOT1560			8	8

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Question Data for Test: 2001 SRO

Question: 249

Peach Bottom Unit 2 has experienced a low reactor level transient due to a leak in the drywell. The TRIP procedures require monitoring and may direct that the use of certain level instrumentation be discontinued as indicated level approaches the lower end of the range.

This direction must be given because actual reactor water level could be:

- [ ] A — above the indicated level due to the effect of instrument reference leg flashing from high drywell temperatures.
- [ ] B — above the reference leg tap with indication downscale due to the effect of excessive water pressure on the reference leg.
- [x] C — below the variable leg tap with indicated level on scale due to the effect of area temperature on instrument calibration.
- [ ] D — below the inidicated level due to the expected instrument error experienced at the top and bottom third of scale.

Explanation of Answer

A. Incorrect - Condition of saturation in reference leg causes level to indicate > actual or upscale.
B. Incorrect - Level above reference would put indication on top of scale.
C. Correct - Off calibration temp will cause indicated level to be above actual - if level then drops indication would bottom out (max delta P) while still on scale.
D. Incorrect - Normal inaccuracies experience at top and bottom of scale are not evaluated in trip tables.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 4.4 SRO Val: 4.4 55.43 ☑

System:	295031	Reactor Low Water Level
KA Group Num:	EA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	EA2.01	Reactor Water Level Indication

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip/SAMP Curves, Tables and Lim	T-BAS	15	12-13	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102	Table DW/T-1		13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	C	18	8	8

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Question Data for Test: 2001 SRO

Question: 250

Peach Bottom Unit 3 has experienced a transient requiring entry into T-103 "Secondary Containment Control" on high area radiation level. A primary system breach has been verified to be discharging into the reactor building and T-103 directs a GP-4, "Manual Scram". Both temperatures and water levels remain below the Maximum Safe Operating limits.

The basis for performing a GP-4 under these conditions is to ensure:

- [ ] A. the availability of Wide Range Level Instrumentation.
- [ ] B. the habitability of the reactor building for personnel access.
- [ ] C. compliance with the Offsite Dose Calculation Manual (ODCM).
- [x] D. a reduction in the energy being discharged into the Secondary Containment.

Explanation of Answer:

A. Incorrect - Wide Range level instrumentation would be impacted by high temperatures which are still below the MSD limit.
B. Incorrect - Habitability is still not ensured due to the primary system breach.
C. Incorrect - ODCM compliance is assured by mitigation actions in T-104.
D. Correct answer.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.5 SRO Val: 3.6 55.43 ☐

System: 295033 — High Secondary Containment Area Radiation Levels

KA Group Num: EK3 — Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: EK3.02 — Reactor Scram

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
T-103 Bases	T-103	SCC-9	13	11	

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Question Data for Test: 2001 SRO

Question: 251

Peach Bottom Unit 3 is in MODE 3 when a complete loss of instrument air occurs. Which of the following describes the operability status of the Reactor Building/Refuel Floor Ventilation Radiation Monitors to detect a high Secondary Containment Ventilation High Radiation?

Radiation Monitors will be:

- [ ] A Operable
- [x] B Inoperable
- [ ] C Operable as long as a Reactor Building/Refuel Floor Supply Fan is still running.
- [ ] D Inoperable until Standby Gas Treatment is started.

Explanation of Answer

A. Incorrect - No flow past detectors.
B. Correct - Loss inst air causes Reactor Bldg/Refuel Floor Exhaust and Supply dampers to close and associated fans to trip. No flow past the detectors.
C. Incorrect - Requires an associated exhaust fan to be operating.
D. Incorrect - SBGTs suction tap is upstream of detectors. No flow past detector.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.8 SRO Val: 4.2 55.43 ☐

System: 295034 Secondary Containment Ventilation High Radiation

KA Group Num: EA2 Ability to determine and/or interpret the following as they apply to:

KA Detail Num: EA2.01 Ventilation Radiation Levels

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Building HVAC	PLOT5040B	II.E.6.b	28	001	6.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Instrument Air	ON-119	Attachment 1	18	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Hi-Lo Diff Pressure	ARC 317 K-5	Operator Actions	2	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Refueling Area Hi-Lo	ARC 317 L-1	Operator Actions	2	4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Building Ventilation Syste	SO 40B.1.A	3.2	1	9	

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Question Data for Test: 2001 SRO

Question: 252

Peach Bottom Unit 3 is operating at rated power, the following conditions apply:

- A high Reactor Building differential pressure exists due to a steam leak.
- Secondary Containment differential pressure is +1.25" water and rising.

Which of the following describes the INITIAL response of Secondary Containment Ventilation?

- [ ] A Reactor Building and Refuel Floor Exhaust Fans will trip.
- [x] B Reactor Building and Refuel Floor Supply Fans will trip.
- [ ] C Reactor Building Equipment Cell Exhaust Fans will trip.
- [ ] D Reactor Building and Refuel Floor Ventilation isolates.

Explanation of Answer

A. Incorrect - Supply fans trip on high diff pressure.
B. Correct - Supply fans trip on high diff pressure.
C. Incorrect - Supply fans trip on high diff pressure.
D. Correct - An isolation does not occur

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE RO Grp: 3 SRO Grp: 2 RO Val: 3.6 SRO Val: 3.6 55.43 ☐

System:	295035	Secondary Containment High Differential Pressure
KA Group Num:	EA1	Ability to operate and/or monitor the following as they apply to:
KA Detail Num:	EA1.01	Secondary Containment Ventilation System

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Building HVAC	PLOT5040B	II.D.1.b	19	001	4.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Hi-Lo Diff Pressure	ARC 317 K-5	Automatic Actions	1	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Refueling Area Hi-Lo	ARC 317 L-1	Automatic Actions	1	4	

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Question Data for Test: 2001 SRO

Question: 254

Peach Bottom Unit 2 is in T-103 "Secondary Containment Control" due to high water level condition in Secondary Containment. The Reactor has been conservatively scrammed and the Group II/III isolations (from the level shrink) are complete.

As the Control Room Supervisor (CRS), you are currently attempting to determine whether a Primary System is discharging into the Reactor Building. Given the above conditions, evaluate the following and determine which constitutes a primary system discharging into the Reactor Building to determine the required direction in T-103, "Secondary Containment Control".

- ☐ A — Leakage from a pipe flange on the discharge of the Reactor Water Cleanup Non-regenerative Heat Exchanger.
- ☐ B — Steam leakage from a rupture on the piping of the #2 Main Steam stop valve inlet.
- ☐ C — Leakage from a weld crack on the "A" RHR suction piping penetration to the Torus.
- ☑ D — Steam leakage from the Standby Liquid Control Injection line just outboard of the drywell penetration.

Explanation of Answer

A. Incorrect, RWCU is isolated on a complete Group II isolation.
B. Incorrect, #2 MSV is not in the Reactor Building and not a T-103 issue.
C. Incorrect, RHR suction is not a primary system.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
SRO	Comprehension	PBAPS	N/A

KA Information

Tier: E/APE — RO Grp: 2 — SRO Grp: 2 — RO Val: 3.3 — SRO Val: 4.0 — 55.43 ☐

System:	295036	Secondary Containment High Sump/Area Water Level
KA Group Num:	2.4	Emergency Procedure/Plan
KA Detail Num:	2.4.20	Knowledge of operational implications of EOP Warnings/Cautions/and Notes

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met: Minor wording enhancement.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Control -	T-103	Note 25.1	11	12	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIPS	PLOT1560			8	9

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Question Data for Test: 2001 SRO

Question: 255

Peach Bottom Unit 2 is shutdown following a Reactor Scram and Group I Isolation. Reactor level is being maintained using the RCIC System on CST suction.

The "RCIC PUMP ROOM FLOOD" alarm activates. The CRS enters T-103, "Secondary Containment Control" and ARC 222 A-4 "RCIC PUMP ROOM FLOOD". There have not been any unexpected CST level changes.

Which of the following describes the correct actions to take under these conditions?

- ☐ A — Control RPV level with HPCI, shutdown RCIC, conduct investigation of RCIC Room by visual inspection from the HPCI Room to RCIC Room door.
- ☐ B — Control RPV level with RCIC, conduct investigation of RCIC Room by visual inspection from the Reactor Building Sump Room to RCIC Room door.
- ☐ C — Control RPV level with RCIC until RCIC components are affected by flooding. Closely monitor other areas for flooding.
- ☑ D — Control RPV level with HPCI, shutdown RCIC, conduct investigation of RCIC Room by visual inspection from the Reactor Building Sump Room to RCIC Room door.

Explanation of Answer:
RCIC Shutdown - may terminate leak, minimize personnel hazard, operator action #3 of ARC.
Reactor Bldg Sump - RCIC Room Door opens into RCIC Room - T-103, Note #36.
Investigate - required per bases of step SC/L-1 and Note #36.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	

KA Information

Tier: E/APE   RO Grp: 3   SRO Grp: 2   RO Val: 3.0   SRO Val: 3.2   55.43 ☐

System:	295036	Secondary Containment High Sump/Area Water Level
KA Group Num:	EA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	EA2.01	Operability of components within the affected area.

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Control Ba	T-103 Bases	SC/L-1 Note #3	6	12	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RCIC Room Flood Alarm Card	ARC 222 A-4			2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip Procedures LP	PLOT1560	C.4			9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip Procedures LP	PLOT1560	C.4			11

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Question Data for Test: 2001 SRO

Question: 256

An accident requiring a General Emergency declaration has occurred on Peach Bottom Unit 2. Initial Protective Action Recommendations (PAR) have been communicated per ERP-200 "Emergency Director".

What action(s) should be taken regarding PARs if the wind direction shifts from the North sector to the North-North-East sector?

- ☐ A. Maintain current PAR's no update is required with this wind shift until next 30 minute update.
- ☑ B. Communicate newly affected sectors in addition to those in initial GE communications with 15 minutes.
- ☐ C. Communicate newly affected sectors and cancel those in initial GE communication within 15 minutes.
- ☐ D. Communicate evacuation of all remaining sectors not communicated in initial GE communication within 30 minutes.

Explanation of Answer

A. Incorrect - Report required if updated.
B. Correct - PARs must be updated to changing wind direction, etc.
C. Incorrect - Previously affected sectors may already be contaminated.
D. Incorrect - PAR only for affected sectors.

Exam Level	Cognitive Level	Facility	Materials
SRO	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 1 RO Val: 2.8 SRO Val: 3.8 55.43 ☐

System:	295038	High Off-Site Release Rate
KA Group Num:	EK1	Knowledge of the operational implications of the following concepts as they apply to:
KA Detail Num:	EK1.03	Meteorological effects on off-site release.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Classification of Emergencies	ERP-101			21	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Director	ERP-200			16	

100

Question Data for Test: 2001 SRO

Question: 257

During the execution of T-102, "Primary Containment Control", which of the following conditions would require direction be given to initiate Drywell Sprays regardless of whether Adequate Core Cooling is assured?

- ☐ A. To prevent exceeding the Pressure Suppression Pressure Limit.
- ☐ B. To maintain Drywell pressure below the Drywell Spray Initiation Limit.
- ☑ C. To mitigate the consequence of a H2 deflagration.
- ☐ D. To mitigate the consequences of containment overpressurization.

Explanation of Answer

A. Incorrect - Sprays are not used to prevent exceeding this limit.
B. Incorrect - Sprays are utilized prior to exceeding this limit but not regardless of ACC.
C. Correct - See T-102 DW/G-3.9 Bases.
D. Incorrect - If ACC is assured sprays are secured to address containment overpressurization.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.3 SRO Val: 3.9 55.43 ☐

System: 500000 — High Containment Hydrogen Concentrations

KA Group Num: EK1 — Knowledge of the operational implications of the following concepts as they apply to high containment hydrogen concentrations.

KA Detail Num: EK1.01 — Containment Integrity

Question Source Information

Ques Source: 1999 NRC PBAPS Exam Question Source:

Ques Mod Met: Minor wording enhancement.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102 - Bases	DW/G-3.9	3.6	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIPS	PLOT1560			8	9

DRAFT
Ro Written

NRC Report

1

Question Data for Test: 2001 RO

Question: 130

Both units are operating in MODE 1 at full power with no Surveillance Testing or other evolutions in progress. The Shift Manager receives a call indicating that one of the licensed operators has been selected for a random substance screening. The selected operator is currently the Unit 2 Reactor Operator (URO). The testing would require the operator to leave the main control room for approximately 45 minutes.

In accordance with the Nuclear Operations Manual, determine the MINIMUM shift response, if any, required for this condition.

- [ ] A. A temporary relief of the URO by the on-shift Plant Reactor Operator (PRO).
- [x] B. A temporary relief of the URO by the fourth Reactor Operator (4th RO).
- [ ] C. A complete turnover of the URO position to the Plant Reactor Operator (PRO).
- [ ] D. No relief is required since licensed operators are exempt from random substance testing while holding a licensed position.

Explanation of Answer

A. The PRO can not relieve the URO to leave the main control room. Must have a CRS and 3 ROs in the Control Room.
B. Correct answer.
C. A complete turnover is not required since it will be less than 1 hour. Also, the PRO cannot formally hold two shift positions.
D. Licensed operators may be excused from random substance exams only on a case by case basis if the CRS determine that conditions do not support the operator leaving.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 1 SRO Grp: 1 RO Val: 3.0 SRO Val: 3.4 55.43

System:	Generic	
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.3	Knowledge of Shift Turnover Practices

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met	Added appropriate reference to question.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Temporary Relief	NOM-C-4.2	2-4	2-3	0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Chapt	PLOT-1526	11.B.5.b	7	0	1f

Question Data for Test: 2001 RO

Question: 133

While monitoring the process computer at rated power on Unit 2, you note that the Process Computer point for Line "A" Feedwater flow shows 0.8 mlbm/hr while actual flow indicated on FR-2565 is approximately normal at 4.5 mlbm/hr.

With this condition present calculated core thermal power will be _______(1)_______ than indicated power and the calculated margin to thermal limits will be _______(2)_______ than actual.

- ☐ A higher (1), larger (2)
- ☐ B higher (1), smaller (2)
- ☑ C lower (1), larger (2)
- ☐ D lower (1), smaller (2)

Explanation of Answer

A. Incorrect-If feedwater flow indicated higher than actual in PC then TP would be higher.
B. Incorrect-If feedwater flow indicated higher than actual in PC then TP would be higher.
C. Correct-Low indicated feedwater flow would result in lower than actual thermal power calculation resulting in larger calculated margins to limits.
D. Incorrect-If calc pwer is lower than actual it would be further from LCO values.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	N/A

KA Information

Tier: PWGs | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.0 | SRO Val: 3.0 | 55.43 ☑

System:	Generic	Generic
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.19	Ability to use plant computer to obtain and evaluate parametric information on system or component status.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Plant Monitoring System	PLOT-5059K	II.D	40	0	3.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Add/Delete sub, Data For Heat Bal	RT-O-059C-550	Table 1	12	2	

3

Question Data for Test: 2001 RO

Question: 135

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".

In accordance with NOM-C-9.1, "Independent Verification", which of the following describes the required action(s)?

- [x] 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.

Explanation of Answer

A. Correct answer.
B. Independent Verifier not authorized to modify COL steps.
C. Independent Verifier not authorized to "N/A" COL steps.
D. COL is correct, valve position is the problem. A-3 is not required here.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: PWGs | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.4 | SRO Val: 3.3 | 55.43 ☐

System:	Generic	
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.29	Knowledge of how to conduct and verify valve lineups.

Question Source Information

Ques Source: 1998 PBAPS NRC Exam

Question Source:

Ques Mod Met: Updated procedural references.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Independent Verification	NOM-C-9.1	6.3.50	8	0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Chapt	PLOT-1527	11.B.4.a	12	0	1j

4

Question Data for Test: 2001 RO

Question: 136

There is a caution tag affixed to the 20C004A Panel which states:

"Orifice Bypass Valve (MO-2-12-053) should be closed if Reactor Pressure is greater than 200 psig".

This Reactor Water Cleanup (RWCU) caution is to prevent:

- [x] A overpressurizing the downstream piping.
- [ ] B exceeding system flow limits.
- [ ] C resin from sloughing off the RWCU filter demin elements.
- [ ] D water hammer in downstream piping.

Explanation of Answer

Procedure SO 12.1.A-2 caution states:

Restricting orifice RO-2-12-106 maximum flow rate is 185 gpm which prevents overpressurizing downstream piping. IF reactor pressure is equal to OR greater than 200 psig, THEN MO-2-12-053, "RWCU Orifice Bypass", should NOT be opened. IF reactor pressure is less than 200 psig THEN MO-2-12-053 may be opened to achieve high flows.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: PWGs RO Grp: 1 SRO Grp: 1 RO Val: 3.4 SRO Val: 3.8 55.43 ☑

System:	Generic	Generic
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.32	Ability to explain and apply system limits and precautions.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Water Cleanup System St	SO 12.1.A-2	Caution	9	28	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Water Cleanup System	PLOT5012	II.D.4	15	1	4.a

5

Question Data for Test: 2001 RO

Question: 139

Which one of the following revisions may be prepared and processed as a Temporary Change (TC) in accordance with Administrative Procedure A-3, "Temporary Changes".

The revision of:

- ☐ A — a continuing action step in an Emergency Response Procedure (ERP)
- ☐ B — the acceptance criteria for a Surveillance Test (ST) procedure
- ☐ C — the wording of a CAUTION statement in a System Operating (SO) procedure
- ☑ D — the stated automatic actions of an Annunciator Response Card (ARC)

Explanation of Answer

A. Changes to ERPs specifically excluded by Exhibit A-3-1.
B. Changes to ST Acceptance Criteria are a change of intent per A-3-1.
C. Changes to NOTES and CAUTION statements are not considered temporary changes. Initiate PPIS per Exh. A-3-1.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	

KA Information

Tier: PWGs — RO Grp: 2 — SRO Grp: 2 — RO Val: 2.5 — SRO Val: 3.4 — 55.43 ☑

System:	Generic	Generic
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.11	Knowledge of the process for controlling temporary changes.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Temporary Change Procedure	A-3	All	All	18	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
A Procedure Lesson Plan	PLOT-1570	B.1.1.b		15	1,2

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

Question Data for Test: 2001 RO

6

Question: 140

Surveillance testing of the 2B Standby Liquid Control (SLC) pump is in progress. The attached Surveillance Test (ST) is complete through step 6.3.20.

Complete the remaining surveillance test steps to determine whether steps 6.3.22 and 6.3.23 are SAT or UNSAT, and the appropriate action(s).

- ☐ A — Step 6.3.22 and 6.3.23 are SAT. Continue with the ST.
- ☑ B — Step 6.3.22 is SAT. Step 6.3.23 is UNSAT. Stop the ST and notify Shift Management.
- ☐ C — Step 6.3.22 is UNSAT. Step 6.3.23 is SAT. Note the UNSAT step in the remarks section and continue with the ST.
- ☐ D — Step 6.3.22 and 6.3.23 are UNSAT. Stop the ST and notify Shift Management.

Explanation of Answer:

Using formula in Step 5, flow rate is 59.7 gpm.
Step 6.3.22 - >43 gpm AND >1255 psig = SAT.
Step 6.3.23 - Result >59.3 gpm = UNSAT.
When a "Black Box" step is marked UNSAT, the operator is required to stop the ST and notify Shift Management.

Exam Level	Cognitive Level	Facility	Materials
RO	Application	PBAPS	Attached surveillance test section.

KA Information

Tier: PWGs — RO Grp: 2 — SRO Grp: 2 — RO Val: 3.0 — SRO Val: 3.4 — 55.43 ☐

System:	Generic	Generic
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.12	Knowledge of surveillance procedures.

Question Source Information

Ques Source: New — Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SLC Pump Functional Test	ST-O-011-301-2			13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Admin Proc. Lesson Plan	PLOT1570	B.1.e.7	12	15	17

6.3.17 LOCALLY **START** 2BP040 **AND THROTTLE** HV-2-11-26 as required to obtain a pressure of 1255 (1255-1280) psig as indicated on PI-2-11-053 **AND RECORD** pressure on Data Sheet 4. RO ______

6.3.18 **WHEN** Test Tank level reaches the lower mark on the SBLC Measuring Stick, **START** stopwatch, **THEN MEASURE** the time required to raise Test Tank level to the upper mark on the SBLC Measuring Stick. RO ______

6.3.19 **STOP** 2BP040. RO ______

6.3.20 **RECORD** time required for level change on Data Sheet 4 to one tenth of a second. RO ______

6.3.21 **CALCULATE** 2BP040 flow rate as follows AND RECORD Flow rate on Date Sheet 4:

$$\frac{52.8 \text{ gal} \times 60 \text{ sec/min}}{\text{Step 6.3.20}} = \text{Flow Rate}$$

3168 / ________ sec = ________ gpm ______ ______

DATA SHEET 4
2BP040 IST DATA

PARAMETER	ACTUAL VALUE	ACCEPTABLE RANGE	ALERT RANGE	ACTION RANGE
Time (Seconds)	53.1	N/A	N/A	N/A
FLOW RATE (gpm) (3168/Time)		51.2 to 59.3	< 51.2 to 50.1	< 50.1 or > 59.3
DISCH PRESSURE (psig)	1260	1255-1280	N/A	N/A

6.3.22 **VERIFY** flow recorded in Data Sheet 4 is $\geq$ 43 gpm **AND** pressure is $\geq$ 1255 psig ______ ☐

6.3.23 **VERIFY** pump test data on Data Sheet 4 does **NOT** fall within Action Range. ______ ☐

Question Data for Test: 2001 RO

7

Question: 144

Unit 3 is in MODE 5 with refueling activities in progress.

Which of the following conditions would require the Reactor Operator to notify the Fuel Handling Director to suspend core alterations in accordance with FH-6C, "Core Component Movement - Core Transfers"?

- ☐ A. Shutdown Cooling (SDC) has been removed from service to complete a swap of SDC loops.
- ☐ B. The white rod permissive light on the C05 panel is NOT lit when the refuel platform is over the core with fuel loaded on the main hoist.
- ☑ C. Wide Range Neutron Count Rate doubles when a fifth fuel bundle is seated around the "A" WRNM detector.
- ☐ D. Receipt of the "A Fuel Pool Serv Water Booster Pump Overcurrent" alarm.

Explanation of Answer:

A. Incorrect, notification not required for SDC swaps.
B. Incorrect, White light should extinguish under these conditions FH6C - 10.2.7.
C. Correct, FH6C - 10.2.9 required notification component movement if any WRNM count rate doubles (after the 4th placed around a detector).
D. Incorrect, Loss of Fuel Pool Cooling Service Water Booster pump has no impact on core alterations.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	N/A

KA Information

Tier: PWGs | RO Grp: 2 | SRO Grp: 2 | RO Val: 2.6 | SRO Val: 3.5 | 55.43 ☑

System:	Generic	
KA Group Num:	2.2	Equipment Control
KA Detail Num:	2.2.27	Knowledge of the refueling process.

Question Source Information

Ques Source: 1999 NRC Exam

Question Source:

Ques Mod Met: Minor wording enhancements and changed distractor 'A'

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Component Movement - Core	FH-6C	10.2.9	25	54	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Peach Bottom Refueling Procedure	NLSRO-0763			3	6

Question Data for Test: 2001 RO

8

Question: 145

To return the plant to a stable condition during a transient, Operations personnel need to enter a High Radiation Area that does not have an existing Radiation Work Permit (RWP).

Which of the following will meet the MINIMUM requirements for an Equipment Operator to enter the area.

- [ ] A. Must be accompanied by an Advanced Rad Worker (ARW) qualified individual.
- [ ] B. Entry into the area is not permitted without the Radiation Protection Manager (RPM) permission.
- [x] C. Must be accompanied by a Level II Radiation Protection Technician qualified individual.
- [ ] D. Entry into the area is not permitted until activation of the Emergency Plan.

Explanation of Answer: HP-C-310 in a effort to return the plant to a stable condition a Level II (ANSI 3.1) RP Technician may act in lieu of a formal RWP to assist workers

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: PWGs | RO Grp: 3 | SRO Grp: 3 | RO Val: 2.6 | SRO Val: 3.0 | 55.43 ☑

System: Generic

KA Group Num: 2.3 — Radiation Control

KA Detail Num: 2.3.1 — Knowledge of 10CFR20 and related facility radiation control requirements.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Radiation Work Permits	PLOT-1760	II.C	6	14	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Radiation Work Permit Program	HP-C-310	7.12		3	

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9

Question Data for Test: 2001 RO

Question: 147

Given the following conditions:

- A male, fully qualified radiation worker at Peach Bottom has just returned from 2 weeks of outage support at Three Mile Island (TMI).
- Total Effective Dose Equivalent (TEDE) received at TMI was 150 mrem.
- After a fall at home, the worker had an ankle x-ray estimated at 10 mrem exposure to the ankle.
- This workers' current TEDE from Peach Bottom for 2001 is 75 mrem.

What is the MAXIMUM annual non-emergency Total Effective Dose Equivalent (TEDE) that can be received at Peach Bottom for the remainder of 2001 WITHOUT exceeding the Federal Exposure Limits.

- [ ] A 4765 mrem
- [x] B 4775 mrem
- [ ] C 4850 mrem
- [ ] D 4925 mrem

Explanation of Answer: Federal Exposure Limit is 5000 mrem. 5000 mrem - 150 mrem (TMI) - 75 mrem (PBAPS) = 4775 mrem. The ankle x-ray is not considered to be occupational exposure.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 3 SRO Grp: 3 RO Val: 2.5 SRO Val: 3.1 55.43 ☑

System: Generic

KA Group Num: 2.3 Radiation Control

KA Detail Num: 2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Dosimetry Program	HP-C-106	7.1.10	3	6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Occupational Dose Limits for Adult	10CFR20.1201				

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Radiation Exposure Limits	PLOT-1730			12	2

Question Data for Test: 2001 RO

10

Question: 148

Peach Bottom Unit 2 was operating at full power when the "A" Recirc Pump tripped. The following conditions exist:

Reactor Power 68%
- Calculated Single Loop Core Flow is 45%
- Initial APRM flux noise level 2%
- Final APRM flux noise level 3%
- APRM flux oscillation period 10 seconds

Use the attached "Peach Bottom Power Flow Operation Map" to determine the required Immediate Operator Action, if any, in accordance with OT-112 "Unexpected/Unexplained Change in Core Flow".

- [ ] A. NO immediate actions are required.
- [ ] B. Raise flow Pump "B" Recirc Pump until Region 2 is exited.
- [x] C. Insert ALL GP-9-2 Appendix 1 Table 1 rods.
- [ ] D. Scram the reactor.

Explanation of Answer

A. Incorrect - Immediate Operation Action is to drive Table 1 rods.
B. Incorrect - Raising recirc flow is a follow-up action and would not be completed until all Table 1 rods are inserted.
C. Correct.
D. Incorrect - Power to Flow conditions don't require a scram and a THI condition does not exit as defined in OT-112.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	Power/Flow Map

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 4.3 SRO Val: 4.6 55.43 ✓

System:	Generic	
KA Group Num:	2.4	Emergency Procedures/Plan
KA Detail Num:	2.4.1	Knowledge of EOP entry conditions and immediate action steps.

Question Source Information

Ques Source: New Question Source:

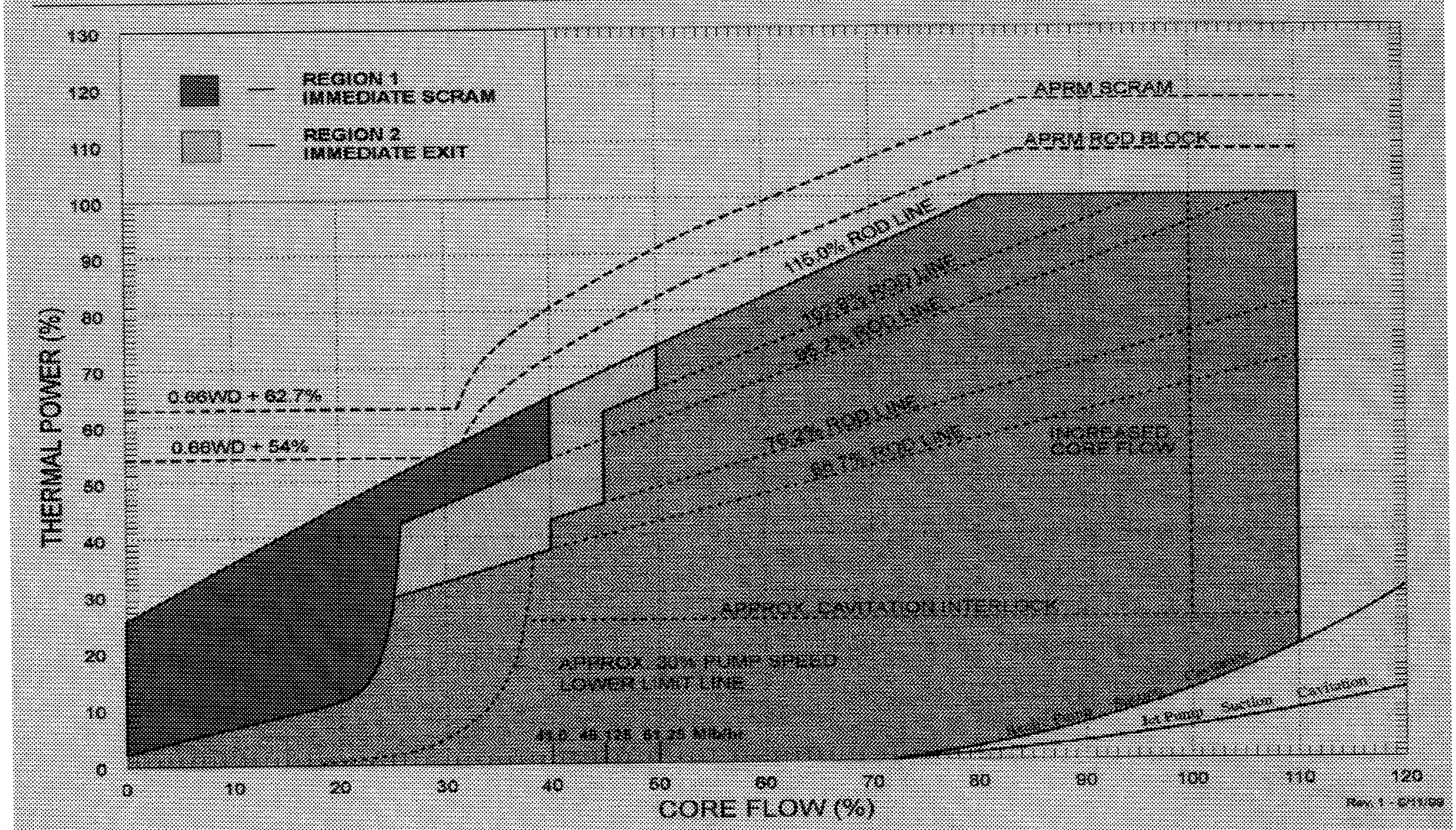
Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Unexpected/Unexplained Change i	OT-112			32	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Procedures	PLOT1540	II.B	6	6	3

PBAPS POWER FLOW OPERATION MAP



CM-1

Question Data for Test: 2001 RO

11

Question: 149

Unit 3 was operating in MODE 1 at 50% power when a plant transient required the crew to scram the unit. The following conditions exist:

- All rods are inserted
- Reactor pressure dropped to 900 psig and has stabilized at approximately 940 psig
- Reactor level dropped to -20", then quickly recovered to its present value of +20" and going up
- HPCI auto started on a valid initiation signal and is injecting into the reactor vessel
- A Main Stack High Radiation Alarm is present
- A Turbine Building 165' elevation Area Radiation Monitor is alarming, reading 6 mr/hr

Select which of the following TRIP procedures should be entered and executed under these conditions.

- ☐ A. Scram Condition (T-100)
- ☑ B. Primary Containment Control (T-102)
- ☐ C. Secondary Containment Control (T-103)
- ☐ D. Radioactive Release (T-104)

Explanation of Answer

A. Incorrect - T-100 should be exited, not executed, due to the T-101 entry condition on 2# DW pressure.
B. Correct - due to Drywell pressure as evidenced by the HPCI auto start.
C. Incorrect - ARM alarm is not in Secondary Containment
D. Incorrect - T-104 is not entered until a High High Main Stack Radiation alarm exists.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 4.0 SRO Val: 4.3 55.43 ✔

System: Generic

KA Group Num: 2.4 — Emergency Procedures / Plan

KA Detail Num: 2.4.4 — Ability to recognize abnormal indications for system operating parameter which are entry level condition.

Question Source Information

Ques Source:	1999 PBAPS NRC Exam	Question Source	
Ques Mod Met	Minor Enhancements		

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102		1	13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TRIP Procedures	PLOT-1560			8	1

Question Data for Test: 2001 RO

12

Question: 150

A steam leak has occurred on Peach Bottom Unit 2. Drywell sprays have been properly placed in service using the "A" RHR Pump. The following conditions exist:

- Torus level 11.5 ft.
- Torus temperature 180 degrees F
- Drywell Bulk Average temperature 260 degrees F
- Drywell pressure is 15 psig and lowering
- Torus pressure is 14 psig and lowering
- "A" RHR Loop flow is 10,000 gpm

Reference the attached T-102 Curves (DW/T-2 and NPSH Curves) as required to select from the following the FIRST point at which Drywell Sprays would need to be terminated.

- ☐ A — Immediately
- ☐ B — When Drywell pressure lowers to 5 psig with Drywell bulk average temperature at 210 degrees F.
- ☑ C — When Torus pressure lowers to 3 psig.
- ☐ D — When Torus pressure lowers to 2 psig.

Explanation of Answer: CAUTION #10 bases say that the operator should maintain NPSH unless directed to use a pump regardless of NPSH concerns. Drywell pressure must stay above 3 psig to maintain NPSH for this RHR flow and Torus temperature.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	T-102 Sheet 3 (NPSH Curves) DW/T-2 from T-102 (DW Spray Initiation Curve)

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 3.3 SRO Val: 3.4 55.43 ☐

System: Generic

KA Group Num: 2.4 — Emergency Procedures/EOPs

KA Detail Num: 2.4.20 — Knowledge of operational implications of EOP warnings, cautions, and notes.

Question Source Information

Ques Source: New

Question Source:

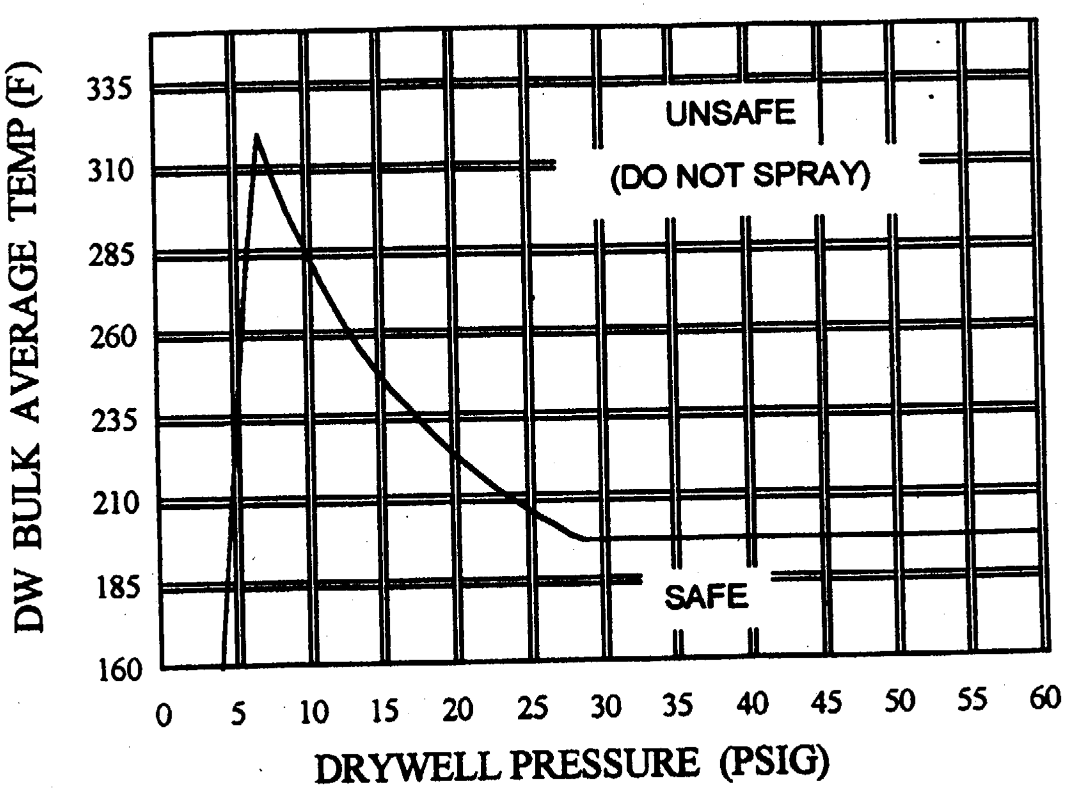
Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control (Bas	T-102 Bases			15	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TRIP Procedures	PLOT1560	II.C	10	15	11

DW SPRAY INITIATION LIMIT



Question Data for Test: 2001 RO

13

Question: 151

SO 48.1.B, "Emergency Cooling Water System Startup" directs the operator to open MO-0841 "Emergency Cooling Water Pump Discharge Valve" by momentarily jumpering between two points in the 00C026D Panel.

This step is accomplished to:

- [ ] A maintain NPSH for the Emergency Cooling Water Pump.
- [ ] B provide an open signal since this valve's normal open circuit has been disabled due to Appendix R concerns.
- [x] C reduce the possibility of water hammer to the ECW piping.
- [ ] D provide a flow path when using the Emergency Service Water Pump in the closed loop lineup.

Explanation of Answer: Precaution 3.3 states that the MO-841 valve must be open when ECW is in closed loop to prevent an ECW System water hammer. A water hammer could occur if the ESW Booster Pumps were to draw a vacuum on the ESW discharge piping AND MO-841 was subsequently opened with the ECW pump already running.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	

KA Information

Tier: PWGs RO Grp: 4 SRO Grp: 4 RO Val: 3.3 SRO Val: 3.7 55.43

System: Generic

KA Group Num: 2.4 Emergency Procedures/EOPs

KA Detail Num: 2.4.24 Knowledge of loss of cooling water procedures.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Cooling Water System	SO 48.1.B	3.3	1	10	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Emergency Cooling Water	PLOT-5048	II.G.2	21	1	1.c

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Question Data for Test: 2001 RO

14

Question: 153

Peach Bottom Unit 2 is operating at 100% power. The CRD Hydraulic System is aligned for normal operation with the "A" Flow Control Valve (AO-2-3-019A) in service.

The 70#-100# air line connection to the air operator for the "A" Flow Control Valve fails, causing the actuator to depressurize.

This loss of air will cause the "A" Flow Control Valve to fail:

- ☐ A — open, resulting in CRD Drive Water Header differential pressure dropping.
- ☐ B — open, resulting in CRD Drive Water Header differential pressure rising.
- ☑ C — closed, resulting in CRD Drive Water Header differential pressure dropping.
- ☐ D — closed, resulting in CRD Drive Water Header differential pressure rising.

Explanation of Answer: The flow control valves will fail closed on loss of air pressure to the actuator. The CRD Drive water differential pressure transmitter senses pressure downstream of the flow control valves. Drive water DP is directly related to the throttled position of the Drive Water Pressure Control Valve (MO-20), and flow from the FCV through MO-20. The loss of flow will cause DP to lower.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 2 RO Val: 3.1 SRO Val: 2.9 55.43 ☐

System: 201001 Control Rod Drive Hydraulic System

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: 201001A10 — CRD Drive Water Header Pressure

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
CRDH System Lesson Plan	PLOT-5003A	E.3.b	27	2	6.C

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Inst. Air	ON-119	Attachment 1	12	14	

Question Data for Test: 2001 RO

15

Question: 154

Peach Bottom Unit 2 was operating at 80% power when the 2A Condensate Pump trips. Reactor level drops to +10" and then stabilizes at +23". Condensate Pump Discharge Header Pressure stabilizes at 350 psig. The Unit Reactor Operator is inserting control rods to reduce reactor power in accordance with GP-9-2, "Fast Power Reduction".

Which of the following correctly describes the effects of this transient on the CRD Hydraulic System?

- [ ] A. CRD System Drive Water Header differential pressure will be higher causing slightly faster control rod speeds.
- [ ] B. CRD System Drive Water Header differential pressure will be lower causing slightly slower control rod speeds.
- [x] C. The CRD Hydraulic System realigns to take a suction from the Condensate Storage Tank.
- [ ] D. Lowered condensate header pressure causes the operating CRD pump to trip on low suction pressure.

Explanation of Answer:

FCV-8031provides 250 gpm from the condensate header to the CRD pump suction and hotwell makeup. If condensate header pressure drops below 430 psig, FCV-8031 automatically closes resulting in CRD pump suctions supplied by the CST.
A & B. Incorrect because CRD FCVs maintain system flow to maintain drive water DP constant.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	N/A

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 2 RO Val: 2.8 SRO Val: 2.8 55.43

System: 201001 Control Rod Drive Hydraulic System

KA Group Num: K6 Knowledge of the effect that a loss or malfunction of the following will have on the system.

KA Detail Num: 201001K60 Condensate System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
CRDH System Lesson Plan	PLOT-5003A	C.1.a	13	2	5.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
P&ID	M-356		Sht 1	62	

Question Data for Test: 2001 RO

16

Question: 156

Peach Bottom Unit 2 is operating at a reduced power for a rod pattern adjustment with the following conditions:

- Core Flow 90%
- Reactor Power 95%
- Both recirculation pumps in service.

Failures cause the A and B Recirculation Pump speeds to rise resulting in a core flow increase to the APRM rod block setpoint.

Use the attached Power Flow Operation Map to determine the expected percent core flow with power at the rod block setpoint? Core flow rose to approximately:

- [ ] A 95%
- [ ] B 102%
- [x] C 108%
- [ ] D 120%

Explanation of Answer

A. Incorrect-Intersection with 100% licensed power.
B. Incorrect-Misinterpreting Rod Line % as flow %.
C. Correct-Following 102.9% Rod Line to intersect with APRM rod block line.
D. Incorrect-Appropriate intersection with APRM Scram Line.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	Exhibit GP-5-1, PBAPS Power Flow Operation Map

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.5 SRO Val: 3.5 55.43

System: 202002 Recirculation Flow Control System

KA Group Num: K3 Knowledge of the effect that a loss or malfunction of the system will have on the following:

KA Detail Num: 01 Core Flow

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met

References

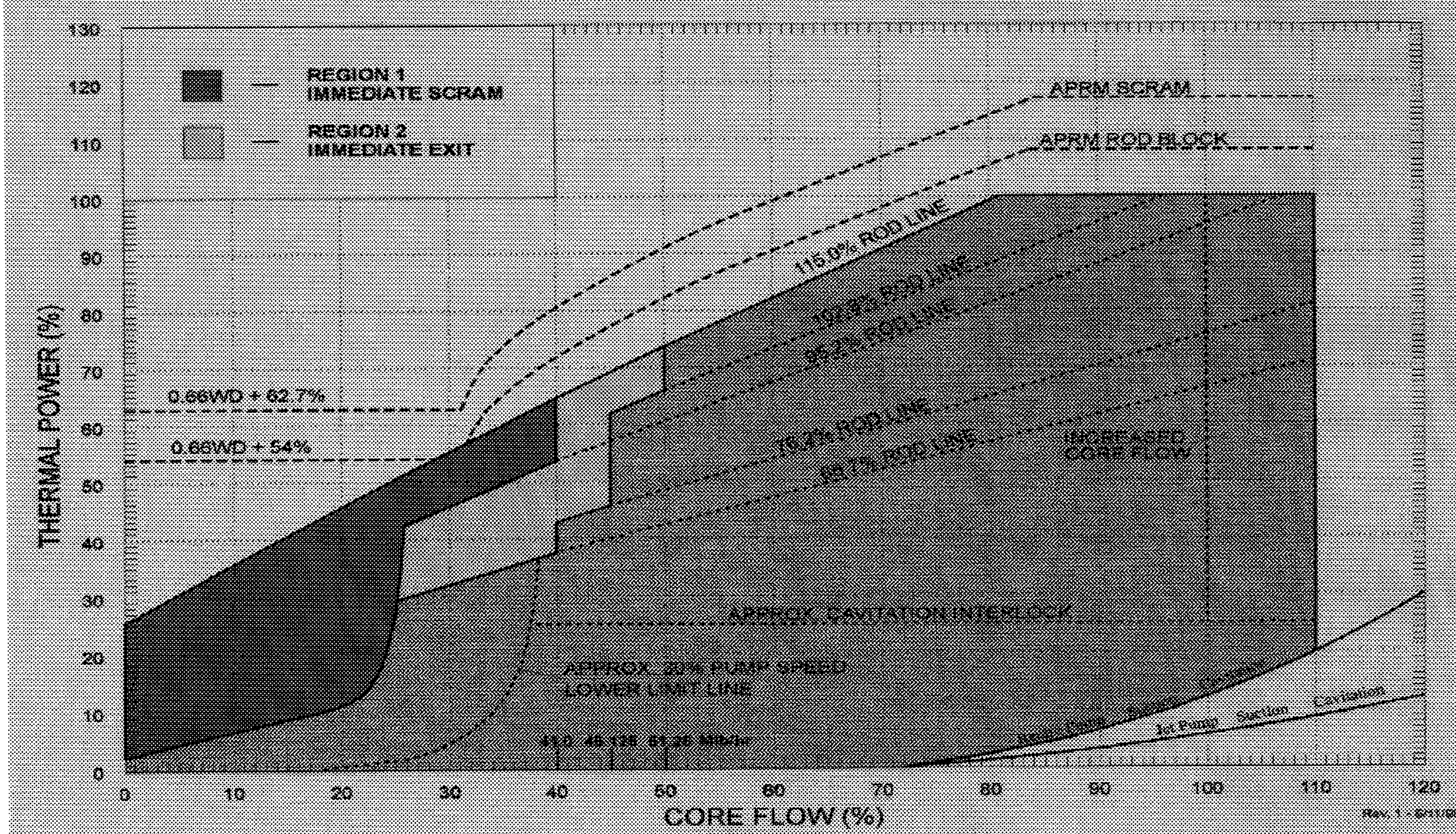
Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Power Flow Operation Ma	Exhibit GP-5-1		1	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT-5002	H	55	1	1.b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT-5002	H	55	1	3.a

Exhibit Gr-5-1
Rev. 2
Page 1 of 2
NHN:nhn

PBAPS POWER FLOW OPERATION MAP



CM-1

Question Data for Test: 2001 RO

17

Question: 158

Both Peach Bottom units were operating at full power with the E-23 4KV bus denergized for troubleshooting. The remainder of the 4KV system was in its normal alignment. The following transient occurs:

- Loss of power to 2SUE.
- E12 Bus fails to transfer to its alternate source.
- E1 Diesel starts and energizes the E12 Bus.
- E43 Bus lock out occurs.

Which one of the following identifies the power supply that would power the 2A and 2D RHR pumps if a LOCA start signal were received on Unit 2?

- [ ] A. Both pumps would be powered from diesel power.
- [x] B. 2A pump would be powered from diesel power, 2D pump would be powered from offsite power.
- [ ] C. 2A pump would be powered from offsite power, 2D pump would not have power.
- [ ] D. Both pumps would be powered from offsite power.

Explanation of Answer: 2A pump powered from E12 which is being powered by the E-1 Diesel.
2D pump powered from E42 which has its off-site source available.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.5 SRO Val: 3.5 55.43

System:	203000	RHR / LPCI: Injection Mode
KA Group Num:	K2	Knowledge of Electricrical Power Suplies to the Following
KA Detail Num:	K2.01	Pumps

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
STBY DG & 4160V Emer. Pwr. Sys	E-8		1 of 2	16	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT-5010	C	19	1	2.a

18

Question Data for Test: 2001 RO

Question: 159

A Dual Unit Design Bases Loss Of Coolant Accident (LOCA) event has occurred on Peach Bottom. The Unit 3 RHR "A" Loop Injection Valve (MO-3-10-25A) normal power suppy (E-134-R-C) has been lost.

Which one of the following describes the expected response of the Unit 3 RHR System with this power supply unavailable?

- [ ] A BOTH loops align for injection with all four RHR pumps running.
- [ ] B ONLY the "B" loop aligns for injection with all four RHR pumps running.
- [x] C BOTH loops align for injection with one RHR pump running in each loop.
- [ ] D ONLY the "B" loop aligns for injection with one RHR pump running in each loop.

Explanation of Answer

A. Incorrect-Only 2 pumps per unit start on dual unit LOCA.
B. Incorrect-Both loop MO25's will align and only 2 pumps start.
C. Correct-MO25 would swap to alternate power and open as designed.
D. Incorrect-Both loop MO25's will align.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 2.5 SRO Val: 2.7 55.43

System:	203000	RHR / LPCI: Injection Mode
KA Group Num:	K2	Knowledge of Electrical Power Supplies to the following
KA Detail Num:	K2.02	Valves

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
E134-R-C MCC or E134 Emer. L.C	AO 56E.2-3	4		7	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Logic	M-1-S-65		48/56	96	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT-5010	C	19	1	2.b

Question Data for Test: 2001 RO

19

Question: 160

The Peach Bottom "A" loop of RHR is in the shutdown cooling mode of operation at 5,000 gpm on the "A" RHR pump when power is inadvertently restored to the pump minimum flow valve (MO-16A).

Which one of the following identifies the response of reactor level as a result of the error?

Reactor level:

- [ ] A. will begin to lower immediately..
- [ ] B. will remain the same regardless of SDC flow rate.
- [ ] C. would begin to lower if shutdown cooling flow is raised.
- [x] D. would begin to lower if shutdown cooling flow is reduced.

Explanation of Answer

A. Min flow closed when flow >500 gpm.
B & C. Valve will open if powered and a low flow condition exists.
C. At ~500 gpm the min flow valve will auto open creating a flow path from the RPV to the torus.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	NA

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 2.8 | SRO Val: 2.9 | 55.43 ☐

System:	205000	Shutdown Cooling System (RHR Shutdown Cooling Mode)
KA Group Num:	K5	Knowledge of the operational implications of the following concepts as they apply to the system.
KA Detail Num:	K5.02	Valve Operation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR SDC Manual Start	SO 10.1.B-2	3.0		24	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT-5010	E	41	1	5.d

Question Data for Test: 2001 RO

20

Question: 161

Peach Bottom Unit 3 has scrammed from 100% power and all Main Steam Isolation Valves are closed. RPV pressure is being maintained 950-1050 psig by manual operation of Safety Relief Valves. RPV level has been restored to +5" using HPCI and RCIC.

The Plant Reactor Operator (PRO) has been directed to place HPCI in the CST-To-CST mode to assist with pressure control. The PRO opens the "COND TANK RETURN" Valve (MO-3-23-24) fully and throttles open the "Full Flow Test" Valve (MO-3-23-21).

Select the answer that correctly describes the HPCI System response as these valves are opened.

HPCI pump:

- [ ] A discharge pressure lowers and speed rises.
- [ ] B discharge pressure and speed both rise.
- [x] C discharge pressure and speed both lower.
- [ ] D discharge pressure rises and speed lowers.

Explanation of Answer: As the test return valves are opened, system pressure decreases due to less flow resistance. System flow goes up. The flow control system closes the governor valve to maintain the flow controller setpoint. This results in speed decreasing.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.5 55.43

System:	206000	High Pressure Coolant Injection System
KA Group Num:	A3	Ability to monitor automatic operations of the system including
KA Detail Num:	A3.01	Turbine Speed: BWR-2, 3, 4

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Rapid Response Card	RRC 23.1-3	C	2	2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Manual Operation	SO 23.1.B-3	4.4	6	13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI System Lesson Plan	PLOT-5023	C.3	14	1	4.I

Question Data for Test: 2001 RO

21

Question: 162

Peach Bottom Unit 2 has experienced a Loss of Off-Site Power (LOOP). The Emergency Diesel Generators have all started and are powering their 4KV busses. Due to a lowering reactor water level, the CRS directs you to use the "Arm and Depress" pushbutton to start the Core Spray system.

After arming and depressing "CS B INITIATION" pushbutton (14A-S10B), what is the expected response of the Core Spray system?

- ☐ A "A", "B" ,"C", and "D" Core Spray pumps start immediately.
- ☐ B "A", "B", "C", and "D" Core Spray pumps start after a time delay.
- ☑ C "B" and "D" Core Spray pumps start immediately.
- ☐ D "B" and "D" Core Spray pumps start after a time delay.

Explanation of Answer: For Core Spray "B" pushbutton starts ONLY "B" & "D" pumps. (On RHR, all 4 start.)
Time delays are not active when the 4 KV busses are powered from the Diesel Generators.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.8 SRO Val: 3.6 55.43

System: 209001 — Low Pressure Core Spray System

KA Group Num: A4 — Ability to manually operate and/or monitor in the control room.

KA Detail Num: A4.05 — Manual Initiation Controls

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Spray	PLOT-5014	V.B.7	18	0	5.h

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Spray Logic	M-1-S-40		2, 3		

32

Question Data for Test: 2001 RO

Question: 163

Peach Bottom Unit 2 is experiencing a Hydraulic Anticipated Transient Without Scram (ATWS) condition. The Standby Liquid Control (SBLC) System has been initiated and the "A" SBLC Pump is injecting into the vessel when a loss of all offsite power condition occurs. All four diesel generators start normally and load their buses.

One minute later, the Reactor Operator is directed to verify the status of the SBLC System. The operator should expect to see the "A" SBLC Pump:

- [x] A running with the squib valve continuity lights lit.
- [ ] B running with the squib valve continuity lights NOT lit.
- [ ] C NOT running with the squib valve continuity lights lit.
- [ ] D NOT running with the squib valve continuity light NOT lit.

Explanation of Answer: The pump should be running since its 480V Emergency Power Supply has been restored by the diesel generators. The squib valve continuity light will be lit even if the valves have fired as long as the pump is running.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 2.9 | SRO Val: 3.1 | 55.43 ☐

System:	211000	Standby Liquid Control System
KA Group Num:	K2	Knowledge of electrical power supplies to the following
KA Detail Num:	K2.01	SBLC Pumps

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Liquid Control System	PLOT-5011	II.D.1.I	14	0	2.a

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Question Data for Test: 2001 RO

Question: 164

Peach Bottom Unit 2 Control Rod Drive (CRD) Drive Water Differential Pressure is indicating abnormally. While investigating, engineering determined that the Standby Liquid Control (SBLC) pipe that provides the reference signal was completely clogged. Engineering stated that any other functions using this piping would be impacted by this condition. The other SBLC pipe is functioning normally.

Determine which of the following would be impacted by this condition.

- ☐ A Jet Pump Differential Pressure indication.
- ☑ B Core Spray Line Break Detection.
- ☐ C Injection of Standby Liquid Control Solution.
- ☐ D Recirculation Loop Flow Indication.

Explanation of Answer

CRD Drive Water dP gets its reference from the SBLC outer pipe.

A. Uses the SBLC inner pipe.
B. Correct
C. Uses the SBLC inner pipe.
D. Uses the recirc flow units, not the SBLC piping.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	NA

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.0 SRO Val: 3.3 55.43 ☐

System: 211000 Standby Liquid Control System

KA Group Num: K1 Knowledge of the physical connections and/or cause-effect relationships between the system and the following

KA Detail Num: K1.01 Core Spray Line Break Detection

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Liquid Control System	PLOT-5011	II.D.5.b	16	0	1.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Boiler Instrumentation P&I	M-352	B-7	1	56	

24

Question Data for Test: 2001 RO

Question: 165

Peach Bottom Unit 2 is operating at 100% power when the 'A' Recirculation Pump Moore Controller Output instantaneously fails to maximum.

Select the expected plant response with no operator action.

- [ ] A. Reactor scram on low RPV level.
- [ ] B. 'A' Recirc Pump trips on high vibration.
- [x] C. Reactor scram on high flux.
- [ ] D. 'A' Recirc MG Set scoop tube locks

Explanation of Answer: Recirc speed rise causes void fraction in the core to drop with an immediate rise in reactor power to the scram setpoint.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 4.4 | SRO Val: 4.4 | 55.43 ☐

System:	212000	Reactor Protection System
KA Group Num:	A3	Ability to monitor automatic operations of the system including
KA Detail Num:	212000A30	Reactor Power

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Protection	PLOT-5060F	C.3.b.D	25	1	1a
Reactor Protection	PLOT5060F	C.3.b.D	25	1	3c

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Question Data for Test: 2001 RO

Question: 166

Peach Bottom Unit 2 is operating at full power with the Traversing In-Core Probe (TIP) System in service. A small steam leak occurs causing drywell pressure to rise to 3.5 psig. The TIP System has continued to operate normally.

Select the statement which describes the expected TIP System response, if any, to these conditions.

The TIP System:

- [ ] A. is responding correctly since it does not automatically isolate under these conditions.
- [x] B. should have isolated. A potential primary containment leakage path exists requiring operator action.
- [ ] C. should have isolated. Operator action is only required if the TIP Room (RB 135') conditions exceed the T-103, Secondary Containment Control Maximum Safe Levels.
- [ ] D. should have isolated. NO operator action is required since the drive cable provides an effective isolation boundary while inserted through the containment penetration.

Explanation of Answer:

A. Determines if candidate understands there is a GP II Isolation for TIP System.
B. Correct
C. Tip should isolate. Immediate Operator Action is required.
D. The cable does go through the containment penetration of concern but does not form a boundary or barrier to the leak path.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 3 | SRO Grp: 3 | RO Val: 3.4 | SRO Val: 3.7 | 55.43

System:	215001	Traversing In-Core Probe
KA Group Num:	A2	Ability to (a) predict the impacts of the following on the system; & (b) based on predictions, use procedures to correct, control, or mitigate the consequences of abnormal conditions or mitigation...
KA Detail Num:	A2.07	Failure to Retract During Accident Conditions

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP System Isol Proc	SO 7F.7.A-2	4.1		2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP Lesson Plan	PLOT5007F	C		0	4a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP Lesson Plan	PLOT5007F	C		0	1e

Question Data for Test: 2001 RO

26

Question: 167

The Traversing In-Core Probe (TIP) System is in operation on Peach Bottom Unit 2 when the following transient occurs:

- Reactor scrammed on low RPV level, HPCI automatically initiated and reactor level has been restored to +30 inches.
- All rods are full in.
- Drywell pressure 1.7 psig and rising slowly.

Which one of the following identifies the expected automatic response, if any, of the TIP System?

The TIP:

- [x] A. retracts and the ball valve closes.
- [ ] B. retracts and the shear valve closes.
- [ ] C. does NOT retract and the TIP purge valve isolates.
- [ ] D. does NOT retract and is unaffected by these conditions.

Explanation of Answer:

A. Correct-TIP will auto reverse and auto isolate the Ball valve due to <1" RPV level.
B. Incorrect-Ball valve auto closes when det. Is in shield not shear valve.
C. Incorrect-TIP retracts
D. Incorrect-TIP retracts

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	NA

KA Information

Tier: SYS | RO Grp: 3 | SRO Grp: 3 | RO Val: 3.3 | SRO Val: 3.4 | 55.43 ☐

System: 215001 — Traversing In-Core Probe

KA Group Num: K1 — Knowledge of the physical connections and/or cause-effect relationships between the system and the following:

KA Detail Num: K1.05 — Primary Containment Isolation System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP iso. In event of CTMT iso.	SO 7F.7.A-2	4.1	1	2	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
TIP	PLOT-5007B	C		0	4.a

Question Data for Test: 2001 RO

27

Question: 168

During a reactor startup on Peach Bottom Unit 3, power is at 40% of rated. Upon selection of the next in sequence rod, an "A" Rod Block Monitor (RBM) fails to successfully complete its Null Sequence.

Which one of the following identifies the impact this failure will have on continued rod withdrawal?

Continued rod withdrawal will be:

- [x] A — prevented due to an INOP trip.on the "A" RBM.
- [ ] B — prevented due to a Comparator trip on the "B" RBM.
- [ ] C — permitted due to an INOP trip on only one of the two RBMs.
- [ ] D — permitted due to only receiving a trouble alarm on "A" RBM.

Explanation of Answer:

A. Correct - A failure to null gives you an INOP trip.
B. Incorrect - Flow Comp. Alarm is due to >10% difference between recirc loop flow indications.
C. Incorrect - Either RBM will generate a rod block.
D. Incorrect - Failure to null will generate an INOP trip.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	NA

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.4 | SRO Val: 3.5 | 55.43 ☐

System:	215002	Rod Block Monitor System
KA Group Num:	K4	Knowledge of this system design feature(s) and/or interlocks which provide for the following.
KA Detail Num:	K4.01	Prevent control rod withdrawal.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC (30C205R)	311 C-3		1	8	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PRNM/OPRM/RBM	PLOT-5060	D	36	1	4.a

Question Data for Test: 2001 RO

28

Question: 169

Peach Bottom Unit 3 is at 50% rated power during a reactor startup when the APRM input to RBM "A" goes downscale. Which one of the following identifies the expected impact on continued rod withdrawal to achieve the desired rod pattern?

Rod withdrawal would be:

- [ ] A prevented until the "A" RBM is manually bypassed.
- [x] B prevented until the APRM is manually bypassed.
- [ ] C permitted due to the "A" RBM being automatically bypassed on APRM failure.
- [ ] D permitted due to the APRM input to the "A" RBM automatically swapping to the backup APRM.

Explanation of Answer

A. Incorrect-This would not remove APRM downscale rod block.
B. Correct-Bypassing APRM removes APRM RB and swaps RBM input to backup APRM.
C. Incorrect-RBM auto bypass occurs however APRM downscale rod block is in.
D. Incorrect-Input APRM swap does not occur until APRM bypass is performed.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	NA

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 3.1 SRO Val: 3.3 55.43 ☐

System: 215002 — Rod Block Monitor System

KA Group Num: A2 — Ability to predict the impacts of the following on the system; and based on those predictions, use procedures to correct control, or mitigate the consequences...

KA Detail Num: A2.03 — Loss of associated reference APRM channel.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 311 C-2	311 C-2		1	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Receipt of Rod Blocks	SO 62.7.A-3	4.0		17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PRNM/OPRM/RBM	PLOT-5060	E	37	1	6.c

Question Data for Test: 2001 RO

29

Question: 170

A reactor startup is in progress on Peach Bottom Unit 2. Just after reaching the point of adding heat, a loss of Uninterruptable AC power (20Y50) occurs. Which one of the following identifies the ability to determine reactor power during this power loss?

Reactor power can:

- ☐ A. be determined on the 20C005 panel ODAs and the 20C036 panel chassis.
- ☐ B. be determined on the 20C005 panel ODAs ONLY.
- ☑ C. be determined on the 20C036 panel chassis ONLY.
- ☐ D. NOT be determined on either the 20C005 panel ODAs or the 20C036 panel chassis.

Explanation of Answer:

A. Incorrect - ODA's lose power.
B. Incorrect - ODA's lose power.
C. Correct - 20Y50 power C005 panel ODA's - 24 VDC powers C03G chassis.
D. Incorrect - Chassis powered from DC.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 2 RO Val: 3.6 SRO Val: 3.6 55.43 ☐

System:	215003	Wide Range Neutron Monitor (WRNM) System
KA Group Num:	K3	Knowledge of the effect that a loss or malfunction of the system will have on the following.
KA Detail Num:	K3.04	Reactor Power Indication

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Uninterruptible AC Power	ON-112-2	2	1	3	

Question Data for Test: 2001 RO

30

Question: 171

A reactor startup is in progress on Peach Bottom Unit 3. Power is on Range 2 of the WRNMs when a loss of power to the "A" WRNM chassis occurs.

Under these conditions, the failure will cause:

- ☐ A — a full reactor scram signal to be generated.
- ☑ B — a rod block and a half scram to be generated.
- ☐ C — only a trouble alarm to be generated.
- ☐ D — the chassis to swap to its alternate power supply.

Explanation of Answer

A. Incorrect - Only one WRNM lost power.
B. Correct - WRNM logic to RPS is 1 out of 2 taken twice with 24 VDC lost all functional outputs from the drawer would occur INOP, short period etc. due to deenergize to trip design.
C. Incorrect - Also a rod block and RPS input.
D. Incorrect - WRNM does not have WLVPS like the APRMs.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 2 RO Val: 2.8 SRO Val: 3.2 55.43 ☐

System: 215003 — Wide Range Nuetron Monitor (WRNM) System

KA Group Num: A2 — Ability to predict the impact of the following on the system, and based on those predictions, use procedures to correct, control or mitigate the consequences. . . .

KA Detail Num: A2.01 — Power supply degraded.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 20C205L G-3	210 G-3		1	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
WRNM	PLOT5060C	D	16	1	4a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
WRNM Schematic	M-1-S-37		13/32	2/2	

31

Question Data for Test: 2001 RO

Question: 172

Peach Bottom Unit 2 is operating at rated power with the recirc flow at 90%. A positive reactivity addition occurs which causes an APRM scram. Recirc flow was constant prior to the scram.

Evaluate this transient and determine the reactor power at which the reactor is expected to scram from an APRM signal.

- [ ] A 114.6% Power
- [x] B 117.6% Power
- [ ] C 119.3% Power
- [ ] D 123.8% Power

Explanation of Answer

A. Incorrect - APRM Rod Block Setpoint
B. Correct - This would be the "Clamp" setpoint for the STP flow biased APRM scram
C. Incorrect - This is the Fixed High Neutron Flux Scram setpoint
D. Incorrect - Calculated scram setpoint using the flow biased scram formula (.66W + 64.4) without regard for the "Clamp"

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.7 SRO Val: 3.7 55.43 ☐

System: 215005 Average Power Range Monitor/Local Power Range Monitor System

KA Group Num: K4 Knowledge of the System design feature(s) and/or interlocks which provide for the following.

KA Detail Num: K4.07 Flow biased trip setpoints.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Exhibit GP-5-1	GP-5		1	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PRNM/OPRM/RBM	PLOT5060	C	23	2	4f

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
APRM HIGH	ARC 211 B-2			4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
APRM/OPRM HI HI/INOP	ARC 211 A-3			5	

32

Question Data for Test: 2001 RO

Question: 173

During power ascension on Unit 3, control rods are being withdrawn to achieve the target rod pattern when APRM #3 spikes to 121% indicated flux and remains there due to a drawer malfunction. All other APRMs are responding normally.

Which one of the following identifies the necessary action, if any, to continue rod withdrawal?

- [x] A. Bypassing of ARPM #3 is required to clear the rod block to permit continued rod withdrawal.
- [ ] B. Bypassing of APRM #3 is required to clear the rod block, permit resetting the half scram, and continued rod withdrawal.
- [ ] C. Following scram recovery, bypassing of APRM #3 is required to permit resetting the full scram prior to attempting rod withdrawal.
- [ ] D. NO action is required to permit continued rod withdrawal.

Explanation of Answer:

A. Correct
B. Incorrect - A half scram will NOT occur
C. Incorrect - A full scram will NOT occur
D. Incorrect - The rod block must be cleared by bypassing the APRM

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS   RO Grp: 1   SRO Grp: 1   RO Val: 3.6   SRO Val: 3.7   55.43 ☐

System: 215005 — Average Power Range Monitor/Local Power Range Monitor System

KA Group Num: A2 — Ability to predict the impacts of the following on the system; and based on those predictions, use procedures to correct, control, or mitigate the consequences. . . .

KA Detail Num: A2.02 — Upscale or downscale trips.

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 211 A-3	211 A-3		1	6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PRNM/OPRM/RBM	PLOT5060	D	34	2	4a

Question Data for Test: 2001 RO

33

Question: 174

The PBAPS Unit 3 Reactor Core Isolation Cooling (RCIC) System receives an automatic initiation signal.

Which of the following describes the operationally correct sequence of events for an automatic RCIC Turbine start and speed control?

- [ ] A. Governor ramp generator activates; governor valve closes partially and then reopens at a controlled rate; steam admission valve starts to open; turbine speed rises.
- [x] B. Steam admission valve starts to open; governor ramp generator activates; turbine speed rises; governor valve closes partially and then reopens at a controlled rate.
- [ ] C. Governor ramp generator activates, steam admission valve starts to open; turbine speed rises; governor valve closes partially and then reopens at a controlled rate.
- [ ] D. Steam admission valve starts to open; turbine speed rises; governor valve closed partially initiating the governor ramp generator and then reopens at a controlled rate.

Explanation of Answer

A. Incorrect - The ramp generator is activated by the steam admission valve "not closed" and the governor valve can't reposition until turbine speed rises to turn oil pump.
B. Correct - The steam admission valve actuates the ramp generator, the governor valve is normally open so as turbine speed rises, and oil pressure builds with turbine speed, the governor valve will close partially and then control turbine.
C. Incorrect - The ramp generator is activated by the steam admission valve.
D. Incorrect - The governor valve will not partially close unless the ramp generator is actuated.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 2.7 SRO Val: 2.7 55.43

System: 217000 Reactor Core Isolation Cooling System (RCIC)

KA Group Num: K5 Knowledge of the operational implications of the following concepts as they apply to the system.

KA Detail Num: K5.06 Turbine Operation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RCIC Lesson Plan	PLOT5013	C.2	12	0	5c

Question Data for Test: 2001 RO

34

Question: 175

Peach Bottom Unit 2 was operating at full power when a Group I isolation occurred. The reactor initially failed to scram until all rods were inserted by ARI.

- Reactor pressure spiked to 1275 psig, but is now being controlled at approximately 1040 psig using the SRVs.
- Reactor level initially dropped to -50", but has recovered and is being maintained at approximately 20" using RCIC. HPCI failed due to a governor malfunction.
- Current CST level is 28 feet.
- Primary Containment parameters; Torus temperature = 98 degrees F, Torus Level = 16 ft., Drywell Pressure = 1.6 psig, Drywell Temperature = 143 degrees F.

Under these plant conditions, which of the following statements is correct regarding RCIC CST to CST operations?

RCIC CST-to-CST operation is prevented:

- ☐ A. until the RCIC System automatic initiation signal seal-in is reset.
- ☐ B. until the HPCI System automatic initiation signal seal-in is reset.
- ☐ C. due to the RCIC suction swap to the Torus.
- ☑ D. due to the HPCI suction swap to the Torus.

Explanation of Answer

A. & B. Initiation signals do not seal-in to the valves.
C. RCIC suction does not swap on Torus high level.
D. HPCI suction swap at 15'10", closing "Cond Tank Return" valve (MO-2-23-24) which must be open for RCIC CST to CST.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.4 | SRO Val: 3.3 | 55.43

System:	217000	Reactor Core Isolation Cooling (RCIC)
KA Group Num:	A2	Ability to (a) predict the impacts of the following on the system; and (b) based on predictions, use procedures to correc, control, or mitigate. . . .
KA Detail Num:	A2.03	Valve closures.

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RCIC System Manual Operation	SO 13.1.B-2	4.4		6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RCIC Lesson Plan	PLOT5013	E.1	28	0	5

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023			1	6

35

Question Data for Test: 2001 RO

Question: 176

The following initial condition existed following a Loss of Coolant Accident on Unit 3:
- The reactor scrammed on high drywell pressure.
- Drywell pressure is 3.5 psig.
- RPV level is +10 inches and slowly lowering.
- RPV pressure is 900 psig and lowering.
- Torus cooling has been placed in service on Loop "A".

Approximately 10 minutes later the following conditions have changed:

- RPV level is -190 inches and steady.
- RPV pressure 290 psig and lowering.

Under these conditions, Torus Cooling:

- [x] A — will automatically isolate and the "A" Loop will automatically align and inject.
- [ ] B — will automatically isolate and the "A" Loop will NOT automatically align and inject.
- [ ] C — must be manually isolated and the "A" Loop will automatically align and inject.
- [ ] D — must be manually isolated and the "A" Loop will NOT automatically align and inject.

Explanation of Answer:

A. Correct - Initially no LPCI initiation. Signal was present - Once LPCI init. Present Torus Cooling realigns to inject - MO-25A opens if <450 # and injects if < ~306 #.
B. Incorrect - MO-25 will open and inject at 290 psig.
C. Incorrect - Auto closure of valves occur if LPCI initiation present and not overridden (S17 & S18).
D. Incorrect - Auto closure of valves occur if LPCI initiation present and not overridden (S17 & S18).

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	None

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.1 | SRO Val: 3.2 | 55.43

System:	219000	RHR/LPCI: Torus/Suppression Pool Cooling Mode
KA Group Num:	A2	Ability to predict impacts of the following in the system; and based on those predictions, use procedures to correct, control or mitigate the consequences. . . .
KA Detail Num:	A2.04	Valve openings

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Electrical Schematic RHR	M-1-S-65		5/14	95	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010	D	33/34	2	5d

Question Data for Test: 2001 RO

36

Question: 177

During a Design Bases Loss of Coolant Accident, which one of the following limits the maximum differential pressure between the drywell and torus air space (torus pressure above drywell pressure) to within design limits?

- [ ] A. Termination of Drywell Sprays.
- [x] B. Operation of Torus to Drywell vacuum breakers.
- [ ] C. Termination of Torus Sprays.
- [ ] D. Operation of Reactor Building to Torus vacuum breakers.

Explanation of Answer:
A. Incorrect - Limits Drywell Pressure to within design.
B. Correct - Vac Bkr's open to redistribute N2 into Drywell Limiting - pressure.
C. Incorrect - Limits Torus Pressure to within design.
D. Incorrect - Limits Torus Pressure to within design.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	None

KA Information

Tier: SYS　RO Grp: 1　SRO Grp: 1　RO Val: 2.9　SRO Val: 3.1　55.43 ☐

System: 223001 Primary Containment System and Auxiliaries

KA Group Num: K4 — Knowledge of System design features and/or interlocks which provide for the following.

KA Detail Num: K4.05 — Maintains proper suppression pool to Drywell differential pressure.

Question Source Information

Ques Source: New　Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment	PLOT5007			0	4f

37

Question Data for Test: 2001 RO

Question: 178

A Refuel Outage is in progress on Peach Bottom Unit 3. The following conditions exist:

- 3C RHR Pump is operating in shutdown cooling.
- Power is lost to the 30Y33 panel due to failure of the Manual Transfer Switch.
- PCIS Shutdown Cooling logic power is lost (due to the loss of 30Y33.)

Which one of the following describes the effect this will have on the operating shutdown cooling loop?

- [x] A — The 3C RHR Pump will trip due to a loss of shutdown cooling suction path.
- [ ] B — The 3C RHR Pump will be damaged due to operating at low flows with no minimum flow protection.
- [ ] C — The RHR Inboard Injection Valve (MO-3-10-25A) will close, the 3C RHR Pump will remain running on minimum flow.
- [ ] D — The Shutdown Cooling Valves will fail "As Is" on the loss of logic power, shutdown cooling will remain in service.

Explanation of Answer:

A. Correct - MO-17 and 18 will close causing the "C" RHR pump to trip on a loss of suction path.
B. Incorrect - The pump will trip.
C. Incorrect - MO-25 will close, but the pump will trip on a loss of suction.
D. Incorrect - MO-17, 18 and 25A will close, pump will trip.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS — RO Grp: 1 — SRO Grp: 1 — RO Val: 3.2 — SRO Val: 3.3 — 55.43 [ ]

System: 223002 — Primary Containment Isolation System

KA Group Num: K3 — Knowledge of the effect that a loss or malfunction of the system will have on the following:

KA Detail Num: K3.16 — Shutdown Cooling System / RHR

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Residual Heat Removal	PLOT5010	II.D.1.b.12	26	1	3.0

Question Data for Test: 2001 RO

38

Question: 179

A plant startup is in progress on Peach Bottom Unit 2. The following conditions exist:

- The Rx Mode Switch is in "Startup".
- Two Turbine Bypass Valves are open.
- Reactor pressure is 940 psig and steady.

Which one of the following describes the plant response, if any, when the "PCIS System I Main Steam Line High Flow" pressure transmitter (DPT-2-118A) fails high?

- [x] A. A half Group I Isolation will occur.
- [ ] B. The associated steam lines MSIV will go closed, a reactor scram will occur due to high reactor pressure.
- [ ] C. No effect, this isolation is bypassed with the mode switch out of "Run".
- [ ] D. All the MSIVs will close due to a Group I Isolation.

Explanation of Answer

A. Correct - One out of two taken twice logic.
B. Incorrect - Half isolation will occur, no MSIVs will close.
C. Incorrect - Half isolation will occur, this Group I signal is not mode switch dependant.
D.Incorrect - Half isolation, no MSIVs will close.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 2.8 SRO Val: 2.9 55.43

System: 223002 Primary Containment Isolation

KA Group Num: K6 Knowledge of the effect that a loss or malfunction of the following will have on the system.

KA Detail Num: K6.06 Various Process Instrumentation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PCIS	PLOT5007G	II.A.3.C	9	0	5f

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
System I Mn Stm Line Hi Flow	ARC 227 D-1			2	

39

Question Data for Test: 2001 RO

Question: 180

Which one of the following describes the Residual Heat Removal (RHR) System physical connections to the Torus Spray Header on Peach Bottom Unit 2?

- [x] A Either loop can spray using the single common spray header in the torus.
- [ ] B Either loop can spray using either loop's spray header in the torus.
- [ ] C Each loop can spray using only its associated loop spray header in the torus.
- [ ] D Only the "B" loop can spray using the single spray header in the torus.

Explanation of Answer

A. Correct - Both loops connect to one 100% capacity spray ring in torus.
B. Incorrect - Each loop does not have it's own header.
C. Incorrect - Each loop does not have it's own header.
D. Incorrect - Both loops supply a common spray header.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 1 RO Val: 3.4 SRO Val: 3.6 55.43 ☐

System: 226001 RHR / LPCI Containment Spray System Mode

KA Group Num: K1 Knowledge of the physical connections and/or cause - effect relationships between the system and:

KA Detail Num: K1.01 Suppression Pool

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Sustem P&ID	M-361		1	75	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010	B	14	2	1b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Initiation of Torus Sprays Using RH	T-204	4	1	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

Question Data for Test: 2001 RO

2/0

Question: 181

A Design Bases Loss of Coolant Accident (LOCA) has occurred on Unit 2 while operating at 100% power. Valve logic failures prevent initiation of torus and drywell sprays. All other systems function normally.

Which one of the following identifies the expected drywell temperature response to this event?

Peak drywell bulk average temeprature:

- [ ] A. Will not exceed 200 degrees F.
- [ ] B. Will not exceed 212 degrees F.
- [x] C. May exceed 281 degrees F.
- [ ] D. May exceed 340 degrees F.

Explanation of Answer

A. Incorrect - 200 degrees F is ~2 psig in DW.
B. Incorrect - 212 is the boiling point of water
C. Correct - Without sprays peak DW bulk ave reaches +295 degrees F DBA
D. Incorrect - This is the temperature for a small break LOCA

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 1 | RO Val: 3.5 | SRO Val: 3.5 | 55.43 ☐

System: 226001 — RHR / LPCI: Containment Spray System Mode

KA Group Num: K3 — Knowledge of the effect that a loss of malfunction of the system will have on the following:

KA Detail Num: K3.02 — Containment / Drywell / Suppression Chamber Temperature.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control Base	T-102 Bases	DW/T-13	23	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560			8	9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DBA LOCA	PLOT1670	D	7	6	3

41

Question Data for Test: 2001 RO

Question: 182

A Loss of Coolant Accident (LOCA) occurred on Peach Bottom Unit 3. The following conditions existed:

- The 'A' Loop of RHR was blocked
- Reactor Pressure was 850 psig and slowly lowering
- Drywell Pressure was 5 psig and slowly rising
- Wide Range Level Transmitter (LT-72B) indicated downscale
- Fuel Zone Level Transmitter (LT-73B) indicated -225" and steady

One minute later:

- The pressure compensator for Fuel Zone Level (PT-404B) fails low
- You are directed to place Torus Spray in service in accordance with T-203-3 "Initiation of Torus Sprays using RHR".

Given these plant conditions, which of the following actions is required to permit Torus Spray to be lined up with the 'B' Loop of RHR.

- [x] A. Place Switches S18 (CTMT Spray Override 2/3 Core Coverage) in the "ON" position and S17 (CTMT Spray Vlv Cont) in manual
- [ ] B. Depress the S33B (Containment Spray Valve Reset) Pushbutton and place S17 (CTMT Spray Vlv Cont) in manual.
- [ ] C. Place Switch S17 (CTMT Spray Vlv Cont) in the "Manual" position
- [ ] D. Depress the S1B (LPCI Lockout Reset) Pushbutton

Explanation of Answer:

A. Correct - Torus logic will see a lower than actual water level, S18 Switch will override this low level condition, S17 will allow diverting RHR flow from the LPCI mode.
B. Incorrect - The S33B is not used until the LOCA is over
C. Incorrect - A loss of pressure compensation would cause torus spray logic to see a lower than actual level. Level would not meet the 2/3 core coverage interlock.
D. Incorrect - The S1B is used to reset the logic after the LOCA signal is clear

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.1 | SRO Val: 3.2 | 55.43

System:	230000	RHR/LPCI: Torus Spray Mode
KA Group Num:	K1	Knowledge of the physical connectins and/or cause-effect relationships between the following:
KA Detail Num:	K1.08	Nuclear Boiler Instrumentation

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Residual Heat Removal	PLOT5010	II.D.1.a.3	26	001	1.j

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Vessel Instrumentation	PLOT5002B	II.C.3.b	18	000	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
P&ID Vessel Instrumentation	M-352		6	38	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Elect Schematic Diagram	M-1-S-65		50	95	

Question Data for Test: 2001 RO

42

Question: 183

During a Loss of Coolant Accident (LOCA) the following plant conditions exist:

- All rods are full in.
- RPV level is -120 inches and rising slowly.
- RPV pressure is 200 psig and lowering slowly.
- Drywell pressure is 8 psig.
- Drywell temperature is 225 degrees F.
- Torus pressure is 5 psig.
- Torus sprays have been initiated but Shift Management has NOT directed their use regardless of Adequate Core Cooling (ACC)

Which one of the following conditions would directly result in an automatic closure of the torus spray valve?

- ☐ A. RPV level drops to -180 inches.
- ☐ B. RPV pressure drops to 0 psig.
- ☑ C. Drywell pressure drops to 0 psig.
- ☐ D. Torus pressure drops to 0 psig.

Explanation of Answer:

A. Incorrect - LPCI Init. Already exists. (DW > 2 psig + < 450 psig RPV).
B. Incorrect - LPCI init. Already exists. (DW > 2 psig + < 450 psig RPV).
C. Correct - If Drywell pressure drops < 1# with LPCI Initiation signal present all spray valves close.
D. Incorrect - Spray logic senses Drywell pressure only.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	None

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 3.4 | SRO Val: 3.3 | 55.43

System: 230000 — RHR / LPCI: Torus / Suppression Pool Spray Mode

KA Group Num: A3 — Ability to monitor automatic operations of the system including:

KA Detail Num: A3.01 — Valve operation.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Electrical Schematic RHR	M-1-S-65				

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010	D	37	0	4s

Question Data for Test: 2001 RO

43

Question: 184

A Refuel Outage is in progress on Peach Bottom Unit 2. The following conditions exist:

- A fuel bundle is currently being transferred from the reactor core to the spent fuel pool.
- The Refuel Bridge has just been positioned over the spent fuel pool.
- The bridge console "Grapple Loaded" light extinguishes.
- The bridge spotter reports the bundle is still grappled.

In accordance with FH-6C, "Core Component Movement - Core Transfers", which of the following actions is required to be taken for the grapple light?

- [ ] A. Return the bundle to its original location in the reactor core, and suspend fuel movement.
- [ ] B. Continue fuel movements and have the Fuel Handling Director (FHD) and Reactor Engineer visually verify the fuel bundle is grappled and loaded.
- [x] C. Suspend fuel movement immediately and contact Reactor Engineering.
- [ ] D. Continue fuel movements and have the Refueling Platform Operator (RPO) and spotter visually verify that the fuel bundle is grappled and loaded.

Explanation of Answer

A. Incorrect - Not conservative, transferring the bundle back over the core with an INOP interlock.
B. Incorrect - FH-6C and Tech Specs requires suspension of core alternations with inoperable interlocks (not to prevent movement to a safe position).
C. Correct
D. Incorrect - FH-6C and Tech Specs requires suspension of core alternations with inoperable interlocks (not to prevent movement to a safe position).

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	

KA Information

Tier: SYS RO Grp: 3 SRO Grp: 2 RO Val: 2.9 SRO Val: 3.8 55.43

System: 234000 Fuel Handling Equipment

KA Group Num: K3 — Knowledge of the effect that a loss or malfunction of the system will have on the following:

KA Detail Num: K3.04 — Core Modifications/Alterations

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Peach Bottom Refueling Procedure	NLSRO0763	III.A.8	10	3	6

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Component Movement - Core	FH-6C	10.2.6	23	52	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Core Component Movement - Core	FH-6C	10.4.6	4	52	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Technical Specifications		3.9.1	3.9-1		

44

Question Data for Test: 2001 RO

Question: 186

A reactor startup is in progress on Unit 2 with the reactor critical on Range 2 of the WRNMs. I&C Testing caused multiple channels of the Main Steam Line (MSL) flow instruments to fail upscale causing a full Group I Isolation on high Main Steam Line (MSL) flow. RPS did not actuate. All other systems responded as designed. As the Reactor Operator, determine if an ATWS is in progress and why.

An ATWS condition:

- ☐ A — exists since a scram should have occurred on high MSL flow.
- ☐ B — exists since a scram should have occurred on MSIV closure.
- ☐ C — does NOT exist since the high MSL flow scram is bypassed with the mode switch NOT in run.
- ☑ D — does NOT exist since the MSIV closure scram is bypassed with the mode switch NOT in run.

Explanation of Answer:

A. Incorrect - There is no high MSL flow scram.
B. Incorrect - The MSIV closure scram is bypassed with the mode switch not in run.
C. Incorrect - There is no high MSL flow scram.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 3 RO Val: 3.1 SRO Val: 3.2 55.43 ☐

System: 239001 Main and Reheat Steam System

KA Group Num: K4 Knowledge of the system design feature(s) and/or interlocks which provide for the following:

KA Detail Num: K4.05 Steam Flow Measurement

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Steam & Pressure Relief	PLOT5001A		47	0	4.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PCIS	PLOT5007G		15	0	1a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.

Question Data for Test: 2001 RO

45

Question: 187

Peach Bottom Unit 3 is performing a plant startup and is currently holding power at 70% while Condensate Demineralizer work is in progress. Which of the following describes the plant response to a Condensate Demineralizer being returned to service prior to being completely filled and vented?

- [x] A — Main Steam Line Radiation Monitors indication will rise.
- [ ] B — Main Stack Radiation Monitors indication will rise.
- [ ] C — Reactor Feedwater Pumps will trip on low suction pressure.
- [ ] D — Main Condenser Vacuum will degrade.

Explanation of Answer

A. Correct - GE Sil 297 states that rad will rise due to additional N-16 production.
B. Incorrect - N-16 radiation will decay before reaching this location.
C. Incorrect - RFP low suction pressure trips have a time delay preventing a trip with even momentary cavitation.
D. Incorrect - The amount of air in one demin is NOT sufficient to impact condenser vacuum after being disbursed in the reactor vessel.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS — RO Grp: 2 — SRO Grp: 3 — RO Val: 3.6 — SRO Val: 3.6 — 55.43 ☐

System:	239001	Main and Reheat Steam System
KA Group Num:	A1	Ability to predict and/or monitor changes in parameters associated with operating the system controls including:
KA Detail Num:	A1.05	Main Steam Line Radiation Monitors

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensate	PLOT5005	II.E.1.C	24	0	5a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
GE SIL	297		1	S1	

Question Data for Test: 2001 RO

46

Question: 188

The following conditions existed at Peach Bottom:

- Both units were operating at full power with no testing in progress.
- Unit 2 Safety Relief Valve "B" has a bellows failure alarm present.

The following event ocurred:

- A small explosion and fast spreading fire erupted when a spark ignited fumes from a can of paint thinner present in the Control Room.
- Both Units were scrammed in accordance with ON-114 "Actual Fire Reported in the Power Block, Diesel Generator Building, Emergency Plan, Inner Screen, or Emergency Cooling Towers". All scram actions were completed.
- The Control Room was evacuated in accordance with SE-10, "Alternative Shutdown".
- The fire has caused the SV-8130A and B, Backup Instrument Nitrogen Valves, to close.

For these conditions, which of the following SRVs are expected to be available for manual pressure control in accordance with SE-10, "Alternative Shutdown"?

- [x] A — ONLY the "A" and "B" SRVs are expected to be available.
- [ ] B — ONLY the "A" and "K" SRVs are expected to be available.
- [ ] C — ONLY the "H" and "E" SRVs are expected to be available.
- [ ] D — ONLY the "H" and "L" SRVs are expected to be available.

Explanation of Answer

A. Correct - the "A" and "B" are fire protected and will have a nitrogen supply via a bypass line that is installed in SE-10.
B. Incorrect - Since bellows failure does not impact manual operation of the "B" SRV. Also, the "K" SRV is not expected to be available in a fire requiring SE-10 entry.
C. Incorrect - The "H" and "E" SRVs can be remotely operated but not during these conditons.
D. Incorrect - The "H" and "L" SRVs can be remotely operated but not during these conditons.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS — RO Grp: 1 — SRO Grp: 1 — RO Val: 3.4 — SRO Val: 3.5 — 55.43

System: 239002 — Relief / Safety Valves

KA Group Num:	K6	Knowledge of the effect that a loss or malfunction of the following will have on the system:
KA Detail Num:	K6.02	Air (Nitrogen) Supply

Question Source Information

Ques Source:	New	Question Source	
Ques Mod Met			

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE-10 Alternative Shutdown	SE-10	Flowchart	1-2	11	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Steam and Pressure Relief	PLOT5001A	II.E.6.f	37-38	1	6i

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE-10 Alternative Shutdown Bases	SE-10	Step ASD/R-18	8	12	

Question Data for Test: 2001 RO

47

Question: 189

OT-114 Bases "Inadvertent Opening of a Relief Valve" directs pulling fuses in a specific order (as listed in Table 1). Which of the following describes this sequence and the bases for this sequence.

Fuses for the:

- [x] A. alternate power supply are pulled first to prevent reactor pressure fluctuations.
- [ ] B. normal power supply are pulled first to prevent reactor pressure fluctuations.
- [ ] C. alternate power supply are pulled first to ensure the white SRV memory light will illuminate when power is restored.
- [ ] D. normal power supply are pulled first, in an attempt to regain control of the SRV from the alternate power supply.

Explanation of Answer

A. Correct
B. Incorrect - Sequence
C. Incorrect - Light indication is lost when fuses are pulled, but will illuminate when fuses are replaced not sequence dependent.
D. Incorrect - Sequence

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	None

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.4 SRO Val: 3.8 55.43 ☑

System:	239002	Relief/Saftey Valves
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.32	Ability to explain and apply system limits and precautions.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Procedures	PLOT1540	II.B.1.c	6	006	4

Question Data for Test: 2001 RO

48

Question: 190

Peach Bottom Unit 2 is operating at 35% power. "Stator Liquid In-Out Hi Temp" AND "Generator Stator Slots Hi Temp" alarms are received. Upon investigation the Plant Reactor Operator determines that an automatic Turbine Generator runback is occurring.

Which of the following describes the appropriate operator response to this condition?

The crew should verify that:

- [ ] A BOTH Recirculation Pumps trip, scram, and enter T-100, "Scram Condition".
- [ ] B "Load Set" runs back continuously until generator current is below 7726 amps. If the runback stops before 7726 amps, perform a GP-4, "Manual Reactor Scram".
- [ ] C "Load Set" runs back until generator current is less than 7726 amps and that bypass valves go full open. Perform GP-9, "Fast Reactor Power Reduction", to prevent a reactor scram.
- [x] D "Load Set" runs back in pulses until generator current is below 7726 amps and that bypass valves open to control reactor pressure.

Explanation of Answer

A. Incorrect - Recirc trips only occur if initial power is above 45%.
B. Incorrect - The main generator runback will reduce power in pulses to 7726 amps which equates to approximately 23% power. A manual scram is not required unless something does not perform properly.
C. Incorrect - The main generator runback will reduce power in pulses to 7726 amps which equates to approximately 23% power. Bypass valves will only have to absorb approximately 17% power which is well within their capability.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 2.8 SRO Val: 2.9 55.43

System: 241000 — Reactor / Turbine Pressure Regulating System

KA Group Num: K5 — Knowledge of the oeprational implications of the following concepts as they apply to the system:

KA Detail Num: K5.05 — Turbine Inlet Pressure vs. Turbine Load

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Stator Cooling	OT-113			6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
EHC Logic Lesson Plan	PLOT 5001DL	C.4	19	0	5c

Question Data for Test: 2001 RO

49

Question: 191

Peach Bottom Unit 2 is operating at full power when the Load Limit Potentiometer output fails causing an output signal of 80%.

Select from the following statements, the one which best represents the plants response to this EHC Logic System failure.

Reactor power and pressure:

- [ ] A will rise resulting in a reactor scram. Condenser vacuum improves.
- [ ] B will lower resulting in a Group I Isolation and reactor scram. Condenser vacuum gets worse.
- [ ] C are stable. Condenser vacuum improves.
- [x] D are stable. Condenser vacuum gets worse.

Explanation of Answer: Control valves would close, bypass valves would open maintaining reactor pressure and power relatively stable. Condenser vacuum gets worse because of the increased energy of the steam put in the condenser from the bypass valves.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS   RO Grp: 1   SRO Grp: 1   RO Val: 3.4   SRO Val: 3.4   55.43 ☐

System: 241000 — Reactor / Turbine Regulating System

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: A1.21 — Main Condenser Vacuum

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Turbine Startup	SO 1B.1.A-2	4.3	4	26	

50

Question Data for Test: 2001 RO

Question: 192

Unit 2 is operating in MODE 1 with the following conditions present:

- Main Generator Load is 1100 Mwe.
- Power factor is .95 lagging.
- Generator hydrogen pressure is 60 psig.

The Power System Director contacts you and requests that you raise reactive loading to 380 MVARs. Use the attached generator capability curve to determine if you can meet this request and what the MAXIMUM reactive loading would be under these conditions.

With the current Main Generator loading, the Power System Director's requested reactive loading of 380 MVARs is:

- [ ] A. NOT acceptable. Maximum reactive loading is 220 MVARs.
- [ ] B. NOT acceptable. Maximum reactive loading is 360 MVARs.
- [x] C. acceptable. Maximum reactive loading is 390 MVARs.
- [ ] D. acceptable. Maximum reactive loading is 590 MVARs.

Explanation of Answer:
A. Incorrect - Used .98 Power Factor Line
B. Incorrect - Used leading versus lagging side of curve
C. Correct
D. Incorrect - Used 75 psig hydrogen pressure curve

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	Unit 2 Main Generator Estimated Capability Curve

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.8 SRO Val: 3.1 55.43 ☑

System:	245000	Main Turbine Generator and Auxiliary Systems
KA Group Num:	2.1	Conduct of Operations
KA Detail Num:	2.1.25	Ability to Interpret Station reference materials such as graphs/nonographs/and tables

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

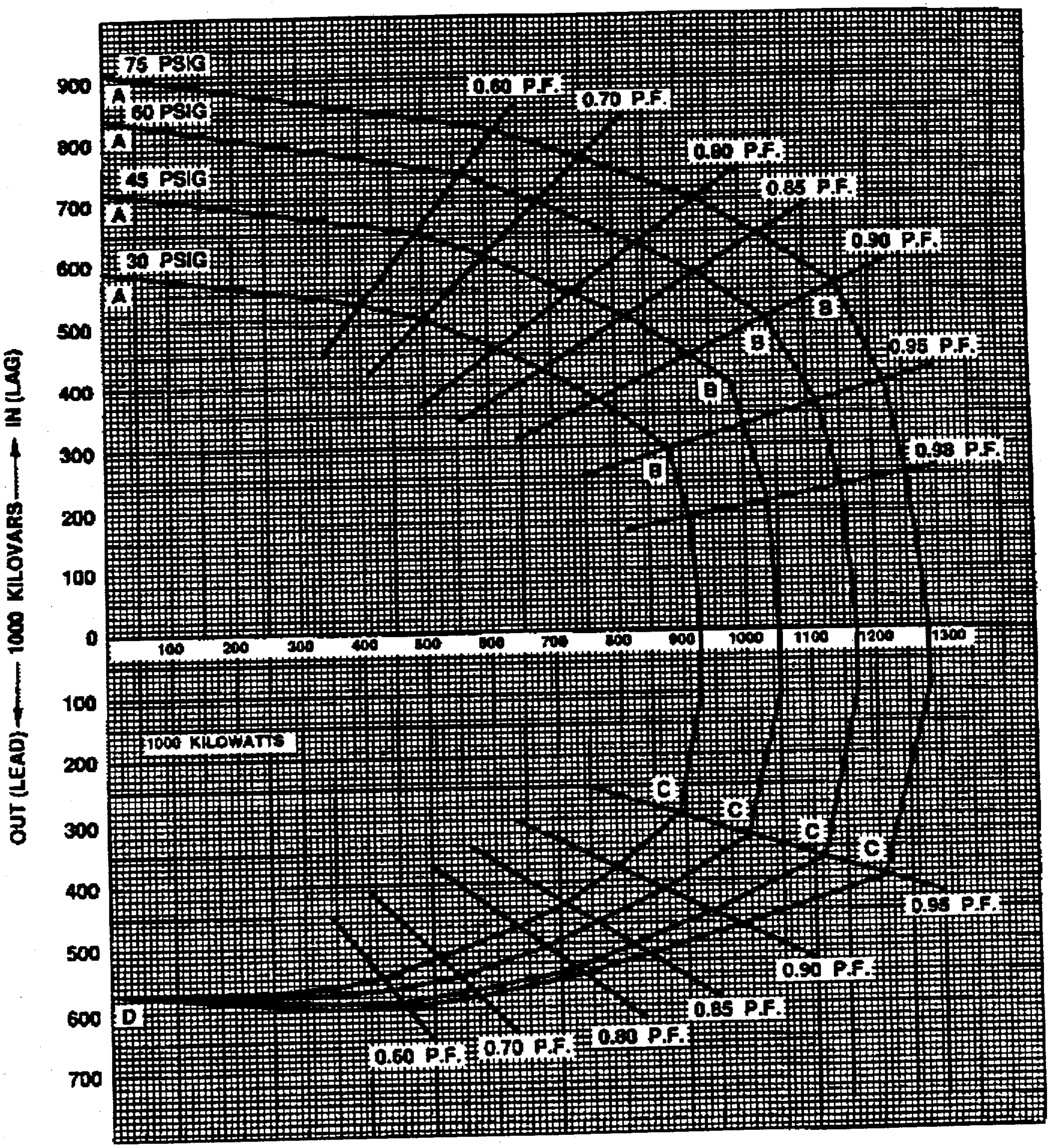
References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Generator Synchronizing and	SO 50.1.A	Figure 1	Last	7	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Main Generator and Auxiliaries	PLOT5050	II.N.1.a	55	2	10

FIGURE 1

ATB 4 POLE 1,280,000 KVA 1800 RPM 22,000 VOLTS
0.90 P.F. 0.60 SCR 75 PSIG HYDROGEN PRESSURE 500 VOLTS EXCITATION



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
LOWER SECTION OF FIGURE 1 OUT (LEAD) IS UNANALYZED FOR OPERATIONS

Question Data for Test: 2001 RO

51

Question: 194

Peach Bottom Unit 2 is operating at 100% power. The "3A HEATER HI LEVEL" annunciator alarms.

Which one of the following describes the design position of the 3A Heater level control valves?

- [x] A Heater drain valve CV-2043A AND dump valve CV-2044A are full open.
- [ ] B Heater drain valve CV-2043A AND dump valve CV-2044A are full closed.
- [ ] C Heater drain valve CV-2043A is full closed AND dump valve CV-2044A is full open.
- [ ] D Heater drain valve CV-2043A is full open AND dump valve CV-2044A is full closed.

Explanation of Answer: Both the dump and drain valve are expected to be full open in order to restore level.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 2 RO Val: 2.9 SRO Val: 3.0 55.43 ☐

System:	259001	Reactor Feedwater System
KA Group Num:	A4	Ability to manually operate and/or monitor in the Control Room:
KA Detail Num:	A4.03	Feedwater Heater / Drain Controls

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
3A Heater Hi Level ARC	ARC 201 D-3		1	2	
Extraction Steam & Drains LP	PLOT5001E	D.2	15	1	5b

52

Question Data for Test: 2001 RO

Question: 195

Peach Bottom Unit 3 is operating at full power when the following actions occur simultaneously:

- RPV level rises.
- H2 and Zinc Injection Systems trip.
- Rod Worth Minimizer activates and indicates less than the Low Power Setpoint (LPSP).
- Both Reactor Recirculation Pumps runback to 30% speed.

Evaluate these conditions and diagnose the cause of this event.

The Digital Feedwater Control System:

- ☐ A Total Steam Flow signal failed high.
- ☐ B Total Steam Flow signal failed low.
- ☐ C Total Feed Flow signal failed high.
- ☑ D Total Feed Flow signal failed low.

Explanation of Answer: Total Feed Flow signal failed low causes 30% RB on <20% Feed Flow; Zinc and H2 Injection use FW flow for power signal. RWM senses Total FW Flow and Total Steam Flow; Either one will activate RWM; decreasing FW Flow signal causes Digital Feedwater to anticipate level decrease and speed up the feed pumps.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	

KA Information

Tier: SYS RO Grp: 1 SRO Grp: 1 RO Val: 3.5 SRO Val: 3.6 55.43 ☐

System: 259002 Reactor Water Level Control System

KA Group Num: K1 Knowledge of the physical connections and/or cause-effect relationships between the systems and the following:

KA Detail Num: K1.04 Reactor Feedwater Flow

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Feedwater LP	PLOT5006	D.2.j	47	1	3

53

Question Data for Test: 2001 RO

Question: 196

A loss of feedwater transient has resulted in an automatic low reactor level scram on Peach Bottom Unit 2.

While verifying automatic actions, the Plant Reactor Operator should expect which of the following Standby Gas Treatment (SBGT) conditions?

- ☐ A All three fans and both filter trains should be in standby.
- ☐ B The "A" fan should have auto started and one filter train should be aligned.
- ☑ C The "A" and "B" fans should have auto started and both filter trains should be aligned.
- ☐ D The "B" and "C" fans should have auto started and both filter trains should be aligned.

Explanation of Answer:
A. Incorrect - Level <1" scram occurred.
B. Incorrect - Two trains intitiate and two fans.
C. Correct - Level <1" scram initiates SBGT A and B for Unit 2 and both filter trains align.
D. Incorrect - This would occur on Unit 3 Low Level event.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS | RO Grp: 1 | SRO Grp: 1 | RO Val: 3.7 | SRO Val: 3.8 | 55.43 ☐

System: 261000 — Standby Gas Treatment System

KA Group Num: K4 — Knowledge of system design feature(s) and/or interlocks which provide for the following:

KA Detail Num: K4.01 — Automatic System Initiation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Low Level	OT-100	4	2	9	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Gas Treatment	PLOT5009A	D		0	4.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Standby Gas Treatment Auto Init.	SO 9A.1.C	4	1	9	

54

Question Data for Test: 2001 RO

Question: 197

PBAPS Unit 2 is operating at 100% power with the electric plant in a normal lineup when the SU-25 breaker trips on low SF6 pressure.

Select the response below which describes the effects on the SU-25 breaker trip on PBAPS Unit 2.

A fast transfer to their alternate sources will occur for 4KV busses:

- [x] A — E12 and E32. A Group II inboard half isolation will be received.
- [ ] B — E22 and E42. A Group II outboard half isolation will be received.
- [ ] C — E12 and E32. The E1 and E3 Emergency Diesels start and run unloaded; a Group II inboard half isolation will be received.
- [ ] D — E22 and E42. The E2 and E4 Emergency Diesels start and run unloaded; a Group II outboard half isolation will be received.

Explanation of Answer:

E22 and E42 normally supplied by 3EA and 343SU, E12 and E32 from SU-2.
DG do not start unless the fast transfer to alternate fails.
Group II outboard half isolation occurs with loss of power to 20Y34, 20Y034 is power from E22.
Group II inboard half isolation occurs with loss of power to 20Y33, 20Y33 is powered from E12.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 1 RO Val: 3.1 SRO Val: 3.4 55.43 ☐

System: 262001 A.C. Electrical Distribution

KA Group Num: A1 — Ability to predict and/or monitor changes in parameters associated with operating the system controls including:

KA Detail Num: A1.01 — Effect on instrumentation and controls of switching power supplies.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
4KV Distribution LP	PLOT5054	D.7,E.1	23,26	2	3g

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
kKV Distribution LP	PLOT5054	D.7,E.1	23,26	2	6

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
GP1, II & III Inbod Half Isol.	GP-8C	2	2	17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
GP I, II & III Outbd Half Isol.	GP-8D	2	2	12	

Question Data for Test: 2001 RO

55

Question: 198

A complete loss of offsite power has occurred at Peach Bottom. No Diesel Generators are available. To minimize the battery discharge rate, SE-11 "Loss of Off Site Power" directs performance of SE-11 Attachment T, "DC Load Shed".

Completion of Attachment T results in deenergization of which of the following circuits?

- [x] A. Alternate Rod Insertion (ARI) Logic
- [ ] B. Reactor Core Isolation Cooling (RCIC) Logic
- [ ] C. Emergency Core Cooling System (ECCS) Logic
- [ ] D. Safety Relief Valve (SRV) Control

Explanation of Answer

A. Correct - ARI is deenergized by Attachment T.
B. Incorrect - Are specifically listed in SE-11 Bases as loads to which Attachment T will maintain power.
C. Incorrect - Are specifically listed in SE-11 Bases as loads to which Attachment T will maintain power.
D. Incorrect - Are specifically listed in SE-11 Bases as loads to which Attachment T will maintain power.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.5 SRO Val: 2.8 55.43 ☐

System:	263000	D.C. Electrical Distribution
KA Group Num:	A1	Ability to predict and/or monitor changes in parameters associated with operating the system controls including:
KA Detail Num:	A1.01	Battery Charging/Discharging Rate

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Off Site Power	SE-11 Bases	Sheet 5	49	11	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DC Load Shed	SE-11 Att.			8	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE Procedures Lesson Plan	PLOT1555			5	9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
SE Procedures Lesson Plan	PLOT1555			5	13

Question Data for Test: 2001 RO

56

Question: 199

While operating at rated conditions with a normal electrical lineup, a piece of scaffolding inadvertently strikes the normal off-site feeder breaker (E-212) for the E-12 bus. This results in an E-212 breaker trip.

The E-1 Emergency Diesel Generator will:

- [ ] A auto start and reenergize the bus.
- [ ] B auto start and the bus will reenergize from the alternate feeder breaker.
- [ ] C NOT auto start and the bus will remain deenergized.
- [x] D NOT auto start and the alternate feeder breaker will reenergize the bus.

Explanation of Answer

A. Incorrect - Only if alternate breaker failed to reenergize the bus.
B. Incorrect - No auto start because a fast transfer will occur.
C. Incorrect - Fast transfer will occur and reenergize the bus.
D. Correct - Alternate breaker closure will reenergize the bus. Diesel will not get a start signal.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: SYS   RO Grp: 1   SRO Grp: 1   RO Val: 3.8   SRO Val: 4.1   55.43 ☐

System: 264000 — Emergency Generators

KA Group Num: K1 — Knowledge of the physical connections and/or cause effect relationships between.

KA Detail Num: K1.01 — A.C. Electrical Distribution

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DG Auto Start and Loading	SO 54.7.E	4		5	

Question Data for Test: 2001 RO

57

Question: 200

The following conditions exist:

- The E-22 4KV Bus has lost power.
- The fast transfer to its alternate off-site source failed
- The E-2 Diesel Generator (DG) started automatically and loaded the E-22 4KV Bus.

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

The E-2 DG is operating in:

- [ ] A. Droop (Parallel), the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes or it will return to the original mode.
- [ ] B. Droop (Parallel), the DG Auto Start Bypass pushbutton must be pressed and synch may be completed without concern for it returning to the original mode.
- [ ] C. Isochronous (Unit), the DG Auto Start Bypass pushbutton must be pressed and synch must be completed within 3 minutes or it will return to the original mode.
- [x] D. Isochronous (Unit), the DG Auto Start Bypass pushbutton must be pressed and synch may be completed without concern for it returning to the original mode.

Explanation of Answer:
A. Incorrect - Will be in the Unit Mode
B. Incorrect - Will be in the Unit Mode.
C. Incorrect - Without a MCA signal, it will not transfer back to the original mode.
D. Correct answer.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	N/A

KA Information

Tier: SYS   RO Grp: 1   SRO Grp: 1   RO Val: 3.7   SRO Val: 3.7   55.43 ☐

System: 264000 Emergency Generators (Diesel/Jet)

KA Group Num: A4 — Ability to manually operate and/or monitor in the control room.

KA Detail Num: A4.04 — Manual start, loading, and stopping of emergency generator: Plant Specific

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Diesel Generators and Auxiliaries	PLOT5052	7	45	0	4

Question Data for Test: 2001 RO

58

Question: 201

Peach Bottom Unit 2 was operating at full power when it experienced a recombiner transient. The PRO reports that he believes that the cause is "Recombiner Process Flashback" to the SJAE after condenser.

Select the following indication that would support this diagnosis.

- ☐ A — A drop in Main Condenser Vacuum (vacuum degrading) due to excess hydrogen and oxygen in the Main Condenser.
- ☑ B — A rise in Air Ejector Discharge Radiation levels on RR-2-17-152 due to a drop in dilution flow.
- ☐ C — A drop in Adsorber Inlet flow on FR-4020 due to reduced offgas flow.
- ☐ D — A rise in Recombiner Delta T on DTR-4025 due to excess hydrogen and oxygen present in the recombiner.

Explanation of Answer:

A. Incorrect - As long as recombination is occurring vacuum will remain steady.
B. Correct - The recombination of H2 & O2 in the after condenser causes less dilution flow at the radiation monitor.
C. Incorrect - Adsorber Inlet flow is based on air inleakage and air inleakage remains steady.
D. Incorrect - The hydrogen and oxygen concentrations have not gone up.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 3.1 SRO Val: 3.1 55.43 ☐

System: 271000 — Offgas System

KA Group Num: K1 — Knowledge of the physical connections and/or cause-effect relationships between the offgas system and the following:

KA Detail Num: K1.01 — Condenser air removal system.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Air Ejector Discharge Radiation Hig	ARC 218 E-2	Cause	1 of 1	7	

Question Data for Test: 2001 RO

58

Question: 202

Peach Bottom Unit 3 is operating at 75% power when it experiences a lowering condenser vacuum. The PRO notes that Off Gas flow is below normal and continues to lower.

Diagnose the potential cause of this lowering Off Gas flow.

- [ ] A — Loss of Main Turbine Steam seal pressure
- [ ] B — Leak in the standby Main Feedwater Pump Recirc Line.
- [ ] C — Loss of Steam Packing Exhauster loop seals.
- [x] D — Loss of Recombiner Jet Compressor steam supply pressure.

Explanation of Answer

A. Incorrect - This will result in air inleakage and a rise in offgas flow.
B. Incorrect - This will result in air inleakage and a rise in offgas flow.
C. Incorrect - This will result in air inleakage thru loop seals and a rise in offgas flow.
D. Correct - This will result in lower suction thru sys. and lower flow.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 3.2 SRO Val: 3.3 55.43

System: 271000 Offgas System

KA Group Num: K6 Knowledge of the effect that a loss or malfunction of the following will have on the system:

KA Detail Num: K6.11 Condenser Vacuum

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensor Low Vacuum Bases	OT-106	Bases	3	18	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
P&ID Offgas Recombiner Sys.	M-331		1	68	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Offgas Recombiner ER Sys.	PLOT5008	E	26	1	6.k

Question Data for Test: 2001 RO

61

Question: 204

The Hydrogen Water Chemistry System uses heat detectors to monitor each of the seven shrouded areas per unit.

These heat detectors are necessary because:

- [ ] A ventilation flow through the shroud makes detection of a fire impossible by any other means.
- [x] B hydrogen burns with an invisible flame and would not be discovered by visual inspection.
- [ ] C heat detection can be used to make an early determination of a hydrogen leak before a fire can occur.
- [ ] D heat detection is the only method available for determining a fire in a shroud.

Explanation of Answer

A. Incorrect - Ventilation is limited in the shroud.
B. Correct
C. Incorrect - Once heat is present a fire has already started.
D. Incorrect - The LEXAN covers will turn black in the presence of heat.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 2 | RO Val: 2.6 | SRO Val: 2.7 | 55.43 ☐

System: 286000 Fire Protection System

KA Group Num: K5 — Knowledge of the operational implications of the following concepts as they apply to Fire Protection System.

KA Detail Num: K5.06 — Heat Detection

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Hydrogen Water Chemistry Ssytem	PLOT5015	II.D.1.I	24	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC Hydrogen Heat Detection	ARC 230 D-2			0	

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Question Data for Test: 2001 RO

62

Question: 205

Peach Bottom Unit 2 is operating at full power when annunciator 216 L-1, "REACT BLDG VENT PANEL 20C132 TROUBLE" alarms. After the Equipment Operator is sent to investigate, he reports that annunciator 20C132 D-2, "EQUIP RM EXH FAN 2A-BV18 STANDBY FAN" is alarming indicating a low delta P across the fan. The 'A' Equipment Cell Exhaust Fan (2A-AV18) fan is blocked for maintenance. The 'B' Equipment Cell Exhaust Fan (2A-BV18) fan is still running but at a lower delta P.

Continued operation with these conditions may:

- [x] A permit the spread of contamination from high to low contamination areas during normal plant operations.
- [ ] B permit the spread of contamination from high to low contamination areas during DBA loss of Coolant Accident conditions.
- [ ] C prevent air inleakage into the reactor building areas of highest contamination during normal plant operations.
- [ ] D prevent air inleakage into the reactor building areas of highest contamination during DBA Loss of Coolant Accident conditions.

Explanation of Answer

A. Correct - Improperly functioning equipment cell fans may permit contamination spread during normal operations.
B. Incorrect - Equipment cell fans trip during accident conditions.
C. Incorrect - Air inleakage is expected into all equipment areas during normal operations.
D. Incorrect - Equipment cell fans trip during accident conditions.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	None

KA Information

Tier: SYS RO Grp: 3 SRO Grp: 3 RO Val: 3.1 SRO Val: 3.2 55.43

System: 288000 Plant Ventilation Systems

KA Group Num: K5 Knowledge of the operational implications of the following concepts as they apply to the system:

KA Detail Num: K5.01 Airborne Contamination Control

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RB HVAC	PLOT5040B	B	10/14	1	5.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
EQUIP RM EXH FAN 2A-BV18 ST	RC 20C132 D-2			1	

Question Data for Test: 2001 RO

63

Question: 206

T-103, "Secondary Containment Control", has been entered on Peach Bottom Unit 2. The following conditions exist:

- The reactor is at rated power and pressure.
- Reactor level is 23 inches.
- Reactor Bldg. 165' General Area radiation level is 9000 MR/HR on ARM #2.11.
- Reactor Bldg. 165' General Area temperature is 150 degrees F on TRS-2-13-139 Point #22.
- Torus Room temperature is 117 degrees F on TRS-2-13-139 Point #'s 8, 9 and 15.
- Annunciator 215 E-2, "REAC BLDG FLOOR DRAIN SUMP HI-HI LEVEL" is annunciating.
- All parameters are rising slowly.

In accordance with T-103, peform a:

- ☐ A GP-3 "Normal Plant Shutdown"
- ☑ B GP-4 "Manual Reactor Scram"
- ☐ C GP-9 "Fast Power Reduction"
- ☐ D T-112 "Emergency Blowdown"

Explanation of Answer

A. Incorrect - Indication of a primary system leak exists.
B. Correct
C. Incorrect - a primary breach exists
D. Incorrect - The same parameter has not exceeded an action level in more than one area.

Exam Level	Cognitive Level	Facility	Materials
Both	Application	PBAPS	T-103 - SCC-4 thru 11 and Tables R1 and T-3.

KA Information

Tier: SYS | RO Grp: 2 | SRO Grp: 1 | RO Val: 3.3 | SRO Val: 3.4 | 55.43 ☐

System:	290001	Secondary Containment
KA Group Num:	A4	Ability to manually operate and/or monitor in the Control Room:
KA Detail Num:	A4.02	Reactor Building Area Temperatures

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Transient Response Implementatio	PLOT1560	II.C	10	008	5

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Control Ba	T-103 Bases	SCC-7	12	12	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment Control	T-103 Flowchart	SCC-7	1	13	

T-103	SH 1 OF 1
SCC	SECONDARY CONTAINMENT CONTROL
PEACH BOTTOM ATOMIC POWER STATION TRIP PROCEDURE	
REV. NO. 14	DATE 10/20/00

ISOLATE ALL SYSTEMS DISCHARGING INTO THE AREA EXCEPT SYSTEMS REQUIRED TO:

- SUPPRESS A FIRE

<u>OR</u>

- BE OPERATED BY THE TRIP PROCEDURES

SCC-4

IS A PRIMARY SYSTEM DISCHARGING INTO THE RX BLDG

#25

SCC-5

(NO OR UNKNOWN)

(YES)

<u>IF</u> WHILE EXECUTING THE FOLLOWING STEPS IT IS DETERMINED THAT A PRIMARY SYSTEM IS DISCHARGING INTO THE RX BLDG,

<u>THEN</u> PROCEED TO STEP SCC-9

#25

SCC-6

M

SCC-9

<u>WHEN</u>

THE SAME PARAMETER EXCEEDS AN ACTION LEVEL IN MORE THAN ONE AREA

<u>AND</u>

A PRIMARY SYSTEM IS <u>NOT</u> KNOWN TO BE DISCHARGING INTO THE RX BLDG,

<u>THEN</u>

CONTINUE

SCC-7

INITIATE A PLANT SHUTDOWN USING GP-3

SCC-8

M

<u>BEFORE</u> ANY PARAMETER EXCEEDS AN ACTION LEVEL, PERFORM THE FOLLOWING:

1. MANUALLY SCRAM THE REACTOR USING GP-4
2. ENTER T-101 AND EXECUTE IT CONCURRENTLY WITH THIS PROCEDURE
3. PERFORM RPV DEPRESSURIZATION PER T-101

SCC-9

T-101

RC-1

<u>WHEN</u>

THE SAME PARAMETER EXCEEDS AN ACTION LEVEL IN MORE THAN ONE AREA

<u>AND</u>

THE PRIMARY SYSTEM BREACH HAS <u>NOT</u> BEEN ISOLATED,

<u>THEN</u>

CONTINUE

SCC-10

PERFORM AN EMERGENCY BLOWDOWN USING T-112

SCC-11

T-112

ES-1

TABLE SC/R-1
RADIATION-ALARM AND ACTION LEVELS

AREA	ALARM LEVEL	ACTION LEVEL (MR/HR)	ARM NUMBER		STATUS
			UNIT 2	UNIT 3	
TORUS ROOM	ALARM SETPT	9×10^3	1.3	5.4	
SUMP ROOM OR RCIC ROOM OR HPCI ROOM	ALARM SETPT ALARM SETPT ALARM SETPT	9×10^3 9×10^3 9×10^3	1.2 1.5 1.4	5.3 5.7 5.6	
A OR C RHR ROOM	ALARM SETPT	9×10^3	1.7	5.9	
B OR D RHR ROOM	ALARM SETPT	9×10^3	1.6	5.8	
A OR C CS ROOM 91'6"/116' EL	ALARM SETPT	9×10^3	1.10	5.10 OR 5.12	
B OR D CS ROOM 91'6"/116' EL	ALARM SETPT	8×10^3	1.8 OR 1.11	6.1	
TIP ROOM	ALARM SETPT	NO ACTION LEVEL	2.8	6.10	
GENERAL AREA 135' EL	ALARM SETPT	9×10^3	2.5, 2.6 OR 2.7	6.7, 6.8 OR 6.9	
RWCU/ISOL VALVE PIT AREA 165' EL	ALARM SETPT	8×10^3	2.12	7.2	
GENERAL AREA 165' EL	ALARM SETPT	9×10^3	2.11	7.1	
GENERAL AREA 195' EL	ALARM SETPT	9×10^3	3.6	7.8	
REFUEL FLOOR	ALARM SETPT	NO ACTION LEVEL	3.7, 3.8, 3.9 OR 3.10	7.9, 7.10, 7.11 OR 7.12	

TABLE SC/T-3
TEMPERATURE-ALARM AND ACTION LEVELS

AREA	ALARM LEVEL (°F)	ACTION LEVEL (°F)	INSTRUMENT TRS-2(3)-13-139 PT # (UNLESS SPECIFIED OTHERWISE)	STATUS
TORUS ROOM	115	135	PT 8,9,14,15,20, OR 24	
RCIC ROOM OR HPCI ROOM	110 110	135 150	PT 2 PT 3	
A RHR ROOM OR C RHR ROOM	110 110	135 135	PT 17 PT 29	
B RHR ROOM OR D RHR ROOM	110 110	135 135	PT 23 PT 6	
A CS ROOM OR C CS ROOM	110 110	135 135	TI-2(3)501 PT 151 TI-2(3)501 PT 152	
B CS ROOM OR D CS ROOM	110 110	135 135	TI-2(3)501 PT 153 TI-2(3)501 PT 154	
STEAM TUNNEL	175	190	PT 1 OR 16	
A ISOL VALVE ROOM (SOUTH)	165	190	PT 12	
B ISOL VALVE ROOM (NORTH)	165	190	PT 18 OR 21	
ISOL VALVE PIT 165' EL	140	150	PT 30	
RWCU REGEN HX ROOM OR A NON REGEN HX ROOM OR B NON REGEN HX ROOM OR A OR B RWCU FLTR DEMIN ROOM OR RWCU BACKWASH VALVE ROOM	160 130 130 115 105	NO ACTION LEVEL	PT 11 PT 28 PT 5 PT 10 OR 27 PT 4	
GENERAL AREA 165' EL (MAY AFFECT RPV LEVEL INST)	105	135	PT 22	

Question Data for Test: 2001 RO

64

Question: 208

Peach Bottom Unit 2 is operating at full power with the Backup Air Compressor is blocked for maintenance. The "B" Instrument Air Compressor trips due to an electrical failure in the compressor motor.

Under these conditions, a complete loss of instrument air would occur upon the loss of:

- ☐ A — ONLY the #1 Aux Bus.
- ☐ B — ONLY the #2 Aux Bus.
- ☑ C — BOTH the #1 and #2 Aux Busses.
- ☐ D — BOTH the E-134 and E-324 Busses.

Explanation of Answer

A. Incorrect - Would only trip the "A" air compressor.
B. Incorrect - Would only trip the "C" air compressor.
C. Correct - Would trip both the "A" and "C" air compressors.
D. Incorrect - The Backup Air Compressor is already blocked.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: SYS RO Grp: 2 SRO Grp: 2 RO Val: 2.8 SRO Val: 2.8 55.43 ☐

System: 300000 — Instrument Air System

KA Group Num: K2 — Knowledge of electrical power supplies to following:

KA Detail Num: K2.01 — Instrument Air Compressors

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Compressed Air System	PLOT5036	II.C.1.d	9	0	2

Question Data for Test: 2001 RO

65

Question: 209

A trip of the "A" Reactor Recirculation Pump has resulted in entry into Region 2 of the Peach Bottom Unit 2 Power/Flow map.

Which one of the following indications would require a manual scram in accordance with OT-112, "Unexpected/Unexplained Change in Core Flow"?

- ☐ A. Greater than a 10% difference between any two APRMs.
- ☑ B. Greater than a 10% difference peak to peak on any APRM.
- ☐ C. LPRM flux noise level rises from 2% to 3%.
- ☐ D. OPRM trip setpoint exceeded on any single APRM.

Explanation of Answer:
A. Incorrect - Similar to 10% diff between any two APRM flow values.
B. Correct - OT-112 THI Indication (2nd bullet).
C. Incorrect - Flux noise must increase by two or more times
D. Incorrect - Requires a trip of two channels of the OPRMs

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.3 SRO Val: 3.4 55.43 ☐

System: 295001 Partial or complete loss of forced core flow circulation.

KA Group Num: AA1 — Ability to operate and/or monitor the following as they apply to:

KA Detail Num: AA1.06 — Neutron Monitoring System

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Unexpected/Unexplained Change i	OT-112	2	1	31	
Recirc/Recirc Flow Control	PLOT5002	J	65		1.b

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Question Data for Test: 2001 RO

66

Question: 210

Peach Bottom Unit 2 is operating in Single Loop with indicated core flow of 50 Mlbm/hr. The Total Core Flow indication must be corrected because it is:

- ☐ A — higher than actual due to reverse flow through the inlet plenum.
- ☑ B — higher than actual due to reverse flow through the idle jet pumps.
- ☐ C — lower than actual due to reverse flow through the inlet plenum.
- ☐ D — lower than actual due to reverse flow through the idle jet pumps.

Explanation of Answer

A. Incorrect - Core flow remains in forward direction with some flow in bottom head reversing through idle Jet Pumps instead of going through core.
B. Correct - 2x idle Jet Pump flow is subtracted from running loop Jet Pump flow.
C. Incorrect - Core flow remains in forward direction with some flow in bottom head reversing through idle Jet Pumps instead of going through core.
D Incorrect - Loop flows are summed resulting in higher than actual core flow.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	

KA Information

Tier: E/APE   RO Grp: 2   SRO Grp: 2   RO Val: 3.1   SRO Val: 3.2   55.43 ☐

System:	295001	Partial or complete loss of forced core flow circulation.
KA Group Num:	AA1	Ability to operate and/or monitor the following as they apply to:
KA Detail Num:	AA1.07	Nuclear Boiler Instrumentation System

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Plant Operation	GP-5	Exhibit GP-5-1	2	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc Sys. Single Loop Operation	AO-2A.1-2	4	3	19	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT5002	D	31	2	1.a

Question Data for Test: 2001 RO

67

Question: 211

Air in-leakage into the Unit 2 Main Condenser has resulted in entering OT-106, "Condenser Low Vacuum".

In accordance with OT-106, direction is given to close the MSIV's in order to prevent:

- ☐ A failure of low pressure turbine last stage buckets.
- ☑ B overpressurizing the main condenser.
- ☐ C excessive main turbine vibration.
- ☐ D overheating the low pressure turbine exhaust hood.

Explanation of Answer

A. Incorrect - The condition for this is vacuum <25 inches and generator load <325 MWE.
B. Correct answer.
C. Incorrect - The concern is for Main Condenser overpressurization.
D. Incorrect - Exhaust hood spray will prevent this.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.6 SRO Val: 3.8 55.43 ☐

System: 295002 — Loss of Main Condenser Vacuum

KA Group Num: AK1 — Knowledge of the operational implications of the following concepts as they apply to:

KA Detail Num: Ak1.03 — Loss of Heat Sink

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
OT-106 Bases	OT-106	3.12	8		

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient LP	LOT1540	II.B.1.c	6	6	4

Question Data for Test: 2001 RO

L8

Question: 213

The following conditions exist at Peach Bottom:

- A loss of all off-site power has occurred.
- The Emergency Diesel Generators (DG) are supplying their respective 4KV switchgear.
- 10 minutes later a failure of the 2A Battery (2AD01) results in the loss of 125VDC to 20D21 supplying the E-1 DG.

Which of the following describes the expected status of the DG for this failure?

- [ ] A The DG will shift to the DROOP (Parallel) mode causing output frequency to drop about 5% to 57 hertz.
- [ ] B The DG engine will trip on mechanical overspeed due to loss of power to the electrical govenor.
- [x] C The DG will continue to run at the previous speed and loading.
- [ ] D The DG voltage will lower due to loss of the exciter field flash supply.

Explanation of Answer: If the D/G is running, all alarms, auxiliary pump starts and automatic trips are lost. However, D/G will not trip due to loss of field and will continue to carry load.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.0 SRO Val: 3.1 55.43 ☐

System: 295004 — Partial or complete loss of DC power.

KA Group Num: AK2 — Knowledge of the interrelations between - and the following:

KA Detail Num: AK2.02 — Batteries

Question Source Information

Ques Source: 1997 PBAPS NRC Exam Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Diesel Generator LP	PLOT5052	D.8.C	55	0	6.g

Question Data for Test: 2001 RO

69

Question: 214

While running the "A" Residual Heat Removal Pump on Unit 2 to support Torus Cooling, a loss of D.C. Control power to the pump breaker occurs.

Under these condtions, if an electrical fault were to occur on the "A" RHR pump motor, the pump would:

- ☐ A trip and could not be restarted until control power was restored.
- ☐ B trip but would restart on a LPCI initiation signal.
- ☐ C continue to run but would trip if the control room switch were placed to trip.
- ☑ D continue to run and could cause the associated bus feeder breaker to trip.

Explanation of Answer

A. Incorrect - Trip coil can not be energized - local manual is only way to operate.
B. Incorrect - Trip coil and 52x closing coil will not energize.
C. Incorrect - Only operation would be local-manual at the breaker.
D. Correct - Loss of DC Control Power will disable fault trips (Trip Coil deenergized) and may result in a fault trip of bus feeder breaker.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.3 SRO Val: 3.4 55.43 ☐

System: 295004 Partial or Complete Loss of DC Power

KA Group Num: AK1 Knowledge of the operational implications of the following concepts as they apply to:

KA Detail Num: AK1.05 Loss of Breaker Protection

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR 4KV Pump Ckt.	E-184				

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010			2	1.f

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
DC Distribution	PLOT5057	E	26	0	3.b

Question Data for Test: 2001 RO

70

Question: 216

Peach Bottom Unit 3 was operating at full power when a spurious Group I Isolation occurred.

Select the statement which describes the expected response of the Reactor Feedpumps during the first minute following this transient.

The Reactor Feedpumps will be:

- [ ] A — providing minimum flow into the reactor due to the reactor level swell caused by the Recirc Pump trips.
- [ ] B — providing minimum flow into the reactor due to the high reactor pressure condition caused by the Group I Isolation.
- [ ] C — attempting to provide maximum flow into the reactor, but will not have sufficient steam to operate due to the Group I Isolation.
- [x] D — providing maximum flow into the reactor due to the reactor level shrink caused by the reactor power drop.

Explanation of Answer

A. Incorrect - The Recirc Pumps don't automatically trip on a Group I Isolation.
B. Incorrect - The shutoff head of the RFPs is above the SRV lift points.
C. Incorrect - There is sufficient steam to run the RFPs for several minutes following a scram.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.8 SRO Val: 3.8 55.43 ☐

System:	295006	Scram
KA Group Num:	AK2	Knowledge of the interrelations between scram and the following:
KA Detail Num:	AK2.02	Reactor Water Level Control System

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
OSPS Reactor Operator Response	NOM-P-10.2:5			0	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Nuclear Operations Manual (Chapt	PLOT1527	IIB.5	12	0	1q

Question Data for Test: 2001 RO

Question: 217

Which one of the following identifies the bases for the 30% Recirculation Pump Speed runback following a scram with RPV level <17 inches?

- [x] A. Limits level shrink to prevent a recirc pump trip.
- [ ] B. Reduces steaming rate to limit loss of inventory.
- [ ] C. Limits power should an ATWS occur during the scram transient.
- [ ] D. Reduces feedwater flow rate to reduce level swell post scram.

Explanation of Answer:

A. Correct - With RCS at rated speed the wide range instruments indicate lower than actual (Jet Pump suction near variable leg tap possibly causing inadvertant ECCS initiations and PCIS Isolation and RCS Pump trip).
B. Incorrect - Bases for low feedwater flow rate 45% runback (loss of FW Pump).
C. Incorrect - Bases for -48" trip function.
D. Incorrect - Bases for FW Limiter.

Exam Level	Cognitive Level	Facility	Materials
RO	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.2 SRO Val: 3.3 55.43 ☐

System: 295006 Scram

KA Group Num: AK3 Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.06 Recirculation pump speed reduction.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 214 B-3	214 B-3		2	13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Recirc/Recirc Flow Control	PLOT5002	E	38	2	4.o

Question Data for Test: 2001 RO

Question: 219

Peach Bottom Unit 2 is being started up in accordance with GP-2, "Normal Plant Startup". The startup has been completed to the point of reactor pressure at 450 psig with 3 Bypass Valves open.

Holding reactor pressure at 450 psig ensures that:

- ☐ A — a sufficient warmup of the feedwater nozzles minimizes the chance of thermal stress cracking.
- ☐ B — the RPV does not exceed 20 degrees F temperature change in a 15 minute interval which corresponds to the administrative limit of 80 degrees F/hr.
- ☐ C — turbine shell warming is monitored and adjusted to maintain turbine first stage pressure below 100 psig.
- ☑ D — a reactor feedpump will be operating prior to the reactor pressure exceeding the condensate pump shutoff head.

Explanation of Answer:
GP-2, NOTE:
"Reactor pressure is held at 450 psig to ensure that a reactor feedpump is operating prior to reactor pressure exceeding the condensate pump shutoff head".

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE   RO Grp: 1   SRO Grp: 1   RO Val: 2.9   SRO Val: 3.2   55.43 ☐

System: 295007 — High Reactor Pressure

KA Group Num: AK1 — Knowledge of the operational implications of the following concepts as they apply to:

KA Detail Num: AK1.01 — Pump Shutoff Head

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Normal Plant Startup Proc	GP-2	Note	65	96	
General Plant Procedures LP	PLOT1530	B	7	11	3

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
General Plant Procedures LP	PLOT1530	B	7	11	4

Question Data for Test: 2001 RO

73

Question: 220

T-101, "RPV Control", Step RC/P-13, provides a list of systems to be used, as necessary, to stabilize RPV pressure below 1050 psig. The list includes the use of HPCI.

Which of the following conditions would prevent the use of HPCI in the "CST to CST Mode" for RPV pressure control?

- [ ] A. Condensate Storage Tank (CST) level indicates 8 feet.
- [ ] B. Torus water level indicates 15 feet.
- [ ] C. Reactor pressure indicates 150 psig.
- [x] D. Reactor water level indicates -51 inches.

Explanation of Answer:

A. Incorrect - Swap to Torus is at 5' 7"
B. Incorrect - Swap to Torus is at 15' 6"
C. Incorrect - Steam pressure isolation is at 75 psig.
D. Correct - Initiation signal at -48 inches RPV level, CST-CST prevented with any initiation signal.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.7 SRO Val: 3.8 55.43 ☐

System: 295007 High Reactor Pressure

KA Group Num: AK3 — Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.02 — HPCI Operation: Plant Specific

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control Bases	T-101B	RC/P-13	24	21	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023	E.1	27	1	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023	C.6 & 7	17	1	4

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
HPCI Lesson Plan	PLOT5023	C.6 & 7	17	1	6

Question Data for Test: 2001 RO

74

Question: 221

Peach Bottom procedure OT-110, "Reactor High Level", requires that for an unexpected rise in level above +46 inches, the operator is to verify that the Main Turbine is tripped.

Under these conditions, OT-110 requires the main turbine trip to be verified to:

- [ ] A reduce the steaming rate to mimize the effects of moisture carryover with the steam.
- [ ] B ensure that all moisture carryover is routed directly to the main condenser.
- [x] C minimize the risk of turbine damage due to moisture carryover with the steam.
- [ ] D eliminate the need to close the MSIVs if reactor water level reaches the bottom of the steam lines.

Explanation of Answer: T.S. Bases states purpose of HILV/Turbine trip is to prevent turbine damage.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.0 SRO Val: 3.2 55.43 ☐

System: 295008 High Reactor Water Level

KA Group Num: AK1 Knowledge of the operational implications of the following concepts as they apply to:

KA Detail Num: AK1.01 Moisture Carryover

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor High Level Bases	OT-110	3.5	6	6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Vessel Lesson Plan	PLOT5004	E.2	38	0	3.d

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Technical Specifications Bases		B.3.3.2.2	3.3-58		

Question Data for Test: 2001 RO

75

Question: 224

T-102, "Primary Containment Control" procedure directs containment venting if pressure is expected to exceed PCPL-A (60 psig).

The bases for venting at this value is to:

- [ ] A. prevent the Hardened Vent line rupture diaphragm from rupturing.
- [ ] B. preserve the structural capability of the Primary Containment hatches.
- [x] C. maintain the ability to operate Safety Relief Valves (SRVs).
- [ ] D. reduce pressure to the safe side of the Drywell Spray Initiation Limit curve.

Explanation of Answer

A. Incorrect - Rupture Diaphragm setpoint is 30 psig.
B. Incorrect - Bases for PCPL-B.
C. Correct - 60 psig is max DW # that N2 press can operate SRVs.
D. Incorrect - Trips do not identify controlling below this curve as strategy.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.8 SRO Val: 4.0 55.43 ☐

System: 295010 High Drywell Pressure

KA Group Num: AK3 Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.01 Drywell Venting

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip Curves, Tables and Limits	T-Bases	22	19-20	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	C	18	8	9

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control Base	T-102 Bases	PC/P-17	16	15	

Question Data for Test: 2001 RO

76

Question: 225

A small steam leak has occurred in the drywell on Peach Bottom Unit 3. All drywell temperature indications are available.

Which of the following describes the proper use of available indications to determine actions for this high drywell temperature transient?

- [x] A — Bulk average temperature indication is used to determine entry into the TRIP procedure.
- [ ] B — Bulk average temperature indication is used to determine RPV Level Instrument availability.
- [ ] C — Highest indicated temperature point is used to determine entry into the TRIP procedure.
- [ ] D — Highest indicated temperature point is used to determine Drywell Spray Limit Curve conditions.

Explanation of Answer:

A. Correct - Bulk average indication or calculated (RT) used for ON and TRIP entry.
B. Incorrect - Points 126/127 used for instrument availability
C. Incorrect - If bulk average and manual calculated average is not available then pt. 136 (Reactor Coolant Pump Area Point) is used.
D. Incorrect - Bulk average is used for saturation curve/Pts 126/127 used for Drywell Spray intiation

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.8 SRO Val: 3.9 55.43 ☑

System: 295012 High Drywell Temperature

KA Group Num: AA2 Ability to determine and/or interpret the following as they apply to:

KA Detail Num: AA2.01 Drywell Temperature

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
T-102 Primary Ctmt Control Bases	T-102 Bases		1	15	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Drywell Temp. Monitoring	RT-O-40C-530			3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIP Procedures	PLOT1560	C	18	8	9

Question Data for Test: 2001 RO

77

Question: 226

Peach Bottom Unit 3 is operating at full power when a Safety Relief Valve fails full open and cannot be reclosed. Torus temperature is 82 degrees F and rising.

Continued torus temperature rise may:

- [ ] A be prevented by placing one loop of Torus Cooling in service.
- [ ] B be prevented by placing both loops of Torus Cooling in service.
- [ ] C NOT be prevented unless power is reduced below 25% regardless of Torus Cooling alignment.
- [x] D NOT be prevented while the plant is at power regardless of Torus Cooling alignment.

Explanation of Answer

A. Incorrect - SRV heat input exceeds Torus Cooling Capacity.
B. Incorrect - SRV heat input exceeds Torus Cooling Capacity.
C. Incorrect - Reducing power will only slightly lower pressure and is accomplished to try to shut the SRV. It does not greatly reduce the heat input of an open SRV.
D. Correct - The plant must be shutdown and depressurzied to prevent the torus from continuing to heat up.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.7 55.43 ☐

System: 295013 High Suppression Pool Temperature

KA Group Num: AK2 Knowledge of the interrelations between- and the following:

KA Detail Num: AK2.01 Suppression Pool Cooling

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Inadvertent Opening of a SRV - Ba	OT-114 Bases	2.0/3.0	1/2	9	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010			1	1.b

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Proc.	PLOT1540	B	6	6	4

Question Data for Test: 2001 RO

78

Question: 227

Peach Bottom Unit 2 was operating at full power when it experieced a loss of feedwater heating. OT-104 "Positive Reactivity Addition" directs total core flow to be reduced. This flow reduction is discontinued when 60 Mlbm/hr flow is reached to PREVENT:

- [ ] A — FLLLP alarms and potential thermal limit violations.
- [ ] B — deep rod insertion from making the power distribution shift more severe.
- [ ] C — exceeding the limits of the "Feedwater Temperature vs. Power" curve in OT-104.
- [x] D — entering Region I or II of the Power to Flow Map.

Explanation of Answer

A. Incorrect - Note states that these alarms may be received during flow reduction.
B. Incorrect - This is the reason for reducing flow not the reason for stopping at 60 Mlbm/hr
C. Incorrect - Stopping the flow reduction will not prevent exceeding this curve.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE   RO Grp: 1   SRO Grp: 1   RO Val: 4.0   SRO Val: 4.1   55.43 ☐

System:	295014	Inadvertent Reactivity Addition
KA Group Num:	AA1	Ability to operate and/or monitor the following as they apply to:
KA Detail Num:	AA1.07	Cold Water Injection

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Positive Reactivity Addition Bases	OT-104 Bases	3.5.2	3	17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Procedures	PLOT1540	11.B.1	6	6	3

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Question Data for Test: 2001 RO

19

Question: 228

A hydraulic ATWS has occurred on Peach Bottom Unit 3. The Control Room Supervisor has directed that control rods be inserted using T-220, "Driving Control Rods During a Failure to Scram".

In accordance with T-220, rods MUST be inserted:

- ☐ A — in any sequence using the Rod Control Switch.
- ☐ B — in the GP-3 shutdown sequence using the Rod Control Switch.
- ☑ C — in any sequence using the Emergency In/Notch Override Switch.
- ☐ D — in the GP-3 shutdown sequence using the Emergency In/Notch Override Switch.

Explanation of Answer:

A. Incorrect - T-220 directs using the Emergency In/Notch Override Switch.
B. Incorrect - T-220 directs using the Emergency In/Notch Override Switch.
C. Correct
D. Incorrect - T-220 states that rods may be inserted in any sequence.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.7 55.43 ☐

System: 295015 — Imcomplete Scram

KA Group Num: AK2 — Knowledge of the interrelations between - and the following:

KA Detail Num: AK2.02 — RMCS

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Driving Rods During Failure to Scr	T-220-2	1	1	3	
PBAPS Trips	PLOT1560	C	17	8	7

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Question Data for Test: 2001 RO

86

Question: 229

The following Peach Bottom Unit 3 conditions exist following a plant transient and scram initiation:

- Rx power is 2 E-2% on WRNMs.
- Rx pressure is 800 psig.
- Rx water level is -90 inches.
- 28 control rods remained at 48.

Under these conditions the reactor is _________(1)_________ and an ATWS ________(2)________.

- [x] A — shutdown (1), does exist (2)
- [ ] B — shutdown (1), does NOT exist (2)
- [ ] C — NOT shutdown (1), does exist (2)
- [ ] D — NOT shutdown (1), does NOT exist (2)

Explanation of Answer:

A. Correct - Power is below 1E0% however all rods are not inserted to 04 or below.
B. Incorrect - Rx will not remain shutdown by rods alone.
C. Incorrect - Power is below 1E0% on WRNM.
D. Incorrect - Power is below 1E0% on WRNM & Rx will not remain shutdown by rods alone.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE — RO Grp: 1 — SRO Grp: 1 — RO Val: 3.6 — SRO Val: 3.7 — 55.43

System:	295015	Incomplete Scram
KA Group Num:	AA1	Ability to operate and/or monitor the following as they apply to:
KA Detail Num:	AA1.07	Neutron Monitoring System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RPV Control	T-101	Notes	1	17	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	B	9	8	8

Question Data for Test: 2001 RO

91

Question: 231

Peach Bottom Unit 3 was operating at 100% power when the "Main Steam Line Hi Radiation" alarm (318 D-2) was received. OT-103, "Main Steam Line High Radiation" was entered.

The OT-103 Immediate Operator Actions direct a GP-9 power reduction to:

- [x] A limit Main Stack release rates to acceptable values.
- [ ] B limit Vent Stack release rates to acceptable values.
- [ ] C reduce the injection rate of the Hydrogen Water Chemistry System.
- [ ] D reduce reactor coolant system conductivity to acceptable values.

Explanation of Answer

A. Correct - OT-103 Bases reduction to below alarm value should not result in unacceptable release rates.
B. Incorrect - No direct release to vent stack
C. Incorrect - Bases for directing hydrogen water chemistry trip if malfunctioning.
D. Incorrect - Coolant conductivity levels would not be reduced by lowering power.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 1 RO Val: 3.6 SRO Val: 3.8 55.43 ☐

System: 295017 — High Off-site Release Rate

KA Group Num: AK3 — Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: AK3.04 — Power Reduction

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
MSL High Radiation - Bases	OT-103	2.1	1	6	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Operational Transient Proc.	PLOT1540	B	6	6	3

Question Data for Test: 2001 RO

82

Question: 232

During an accident condition at Peach Bottom, the PRO notices that the Unit 2 Vent Stack radiation is high. Which one of the following identifies a possible source of this release?

- [ ] A. Recombiner Building Ventilation Exhaust
- [ ] B. Standby Gas Treatment Exhaust
- [ ] C. PEARL Building Ventilation Exhaust
- [x] D. Radwaste Building Ventilation Exhaust

Explanation of Answer:
A. Incorrect - Exhausts to Unit 3 Vent Stack.
B. Incorrect - Exhausts to Main Stack.
C. Inocrrect - Exhausts to Unit 3 Vent Stack.
D. Correct - Exhausts to Unit 2 Vent Stack.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE | RO Grp: 2 | SRO Grp: 1 | RO Val: 3.6 | SRO Val: 4.3 | 55.43 ☑

System:	295017	High Offsite Release Rate
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.04	Source of Offsite Release

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Vent Stack High Rad	ON-104	2.1.1	1	9	
P&ID Prim & Secon.Iso. Control	M-391		2	30	

Question Data for Test: 2001 RO

83

Question: 233

Peach Bottom Unit 3 was operating at full power when the "A" Turbine Building Closed Cooling Water (TBCCW) pump tripped on an electrical fault in the motor. The "B" TBCCW pump is blocked.

Determine the impact on continued power operations.

- ☐ A. A reactor power reduction will be required due to a loss of Main Generator Hydrogen Cooling.
- ☑ B. A reactor power reduction will be required due to a loss of Isophase Bus Cooling.
- ☐ C. An immediate plant shutdown will be required due to a loss of cooling to the CRD Pumps.
- ☐ D. An immediate plant shutdown will be required due to a loss of Instrument Air to the Outboard MSIVs.

Explanation of Answer

A. Incorrect - Hydrogen coolers are supplied by service water.
B. Correct
C. Incorrect - The CRD pumps receive cooling from RBCCW on a loss of TBCCW.
D. Incorrect - The Air Compressors receive cooling from RBCCW on a loss of TBCCW.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.1 SRO Val: 3.2 55.43 ☐

System:	295018	Partial or complete loss of Component Cooling Water
KA Group Num:	AK3	Knowledge of the reasons for the following responses as they apply to partial or complete loss of Component Cooling Water.
KA Detail Num:	AK3.07	Cross-connecting with backup systems.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Turbine Building Closed Co	ON-118 Bases			4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Turbine building Closed Cooling W	PLOT5034	VI.B.2	12	0	3

Question Data for Test: 2001 RO

84

Question: 234

A reactor startup is in progress on Peach Bottom Unit 3. The following conditions exist:

- One condensate pump is in service.
- Three condensate demineralizers are in service.
- The condensate system is lined up for "Long Path Recirc for Startup Level Control"
- CV-2110, "Condensate Recirc Flow" valve is partially open and controlling in automatic.
- The instrument air line to the "Condensate Recirc Flow" valve CV-2110 breaks.

Which of the following describes the impact of this failure?

The CV-2110 "Condensate Recirc Flow" valve fails:

- [ ] A. full closed, causing reactor level to rise.
- [ ] B. full open, causing reactor level to lower.
- [x] C. full closed, causing the condensate pump to overheat.
- [ ] D. full open, causing condensate pump motor damage due to high current.

Explanation of Answer:

This is similar to an actual Peach Bottom Event.
A. Incorrect - level will be controlled automatically by the start up level controller
B. Incorrect - CV-2110 fails closed on a loss of air.
C. Correct
D. Incorrect - CV-2110 fails closed on a loss of air.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.2 SRO Val: 3.2 55.43 ☐

System:	295019	Partial or complete loss of instrument air.
KA Group Num:	AK2	Knowledge of the interrelations between - and the following:
KA Detail Num:	AK2.07	Condensate System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met	

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Condensate	PLOT5005	II.E.2.a.1	23	0	6.a

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Instrument Air	ON-119	Attachment 1	42	14	

Question Data for Test: 2001 RO

85

Question: 235

A loss of instrument air transient occurs on Peach Bottom Unit 3. In accordance with ON-119 "Loss of Instrument Air", the reactor must be scrammed if any rod begins to drift in due to lowering scram pilot air header pressure.

What is the bases for this direction?

- ☐ A — To ensure that the scram discharge volume is fully isolated during the scram.
- ☑ B — To ensure that various scram valve opening pressures do not result in a random rod pattern.
- ☐ C — To ensure that the individual control rod scram inlet valves do not open before the scram outlet valves.
- ☐ D — To ensure that sufficient volume exists in the scram discharge volume to complete a full scram.

Explanation of Answer:

A. Incorrect - Loss of air will result in SDV isolation.
B. Correct - To avoid random rod insertion due to varying scram valve opening pressures.
C. Incorrect - Scram outlet valves open prior to scram inlet valves due to greater spring preload.
D. Incorrect - An automatic scram would be initiated off SDV high level PRIOR to there being insufficient volume in the SDV.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.4 SRO Val: 3.6 55.43 ☑

System:	295019	Partial or Complete Loss of Instrument Air
KA Group Num:	2.4	Emergency Procedure/Plan
KA Detail Num:	2.4.11	Knowledge of abnormal condition procedures.

Question Source Information

Ques Source: 1999 PBAPS NRC Exam Question Source:

Ques Mod Met: Minor wording enhancement.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Instrument Air - Bases	ON-119	Step 2.1 Bases	2		

Question Data for Test: 2001 RO

86

Question: 236

Unit 2 has experienced a loss of shutdown cooling. ON-125 "Loss of Shutdown Cooling" directs you to determine the expected decay heat load using Operator Aid 95-04 located on the back of Panel 20C005A.

The information necessary to determine expected heat load using this Operator Aid is:

- [ ] A current heat up rate.
- [ ] B current WRNM indicated power.
- [ ] C power history before shutdown.
- [x] D elapsed time since shutdown.

Explanation of Answer: The operator aid relates time since shutdown to decay heat load in megawatts.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 3 SRO Grp: 2 RO Val: 3.6 SRO Val: 3.8 55.43 ☐

System: 295021 Loss of Shutdown Cooling

KA Group Num: AK1 Knowledge of the operational implications of the following concepts as they relate to a loss of shutdown cooling.

KA Detail Num: AK1.01 Decay Heat

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Shutdown Cooling - Bases	ON-125	Step 2.8.7	11	1	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Expected Decay Heat Operator Aid	OP Aid 95-04				

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550			7	3

Question Data for Test: 2001 RO

87

Question: 237

The following Control Rod Drive System indications are observed on the Unit 2 C05 panel:

Drive Water dp - 0 psid
Cooling Water dp - 0 psid
Charging Header Pressure - 1015 psig
CRD System Flow - 0 gpm

Select the condition described below which would result in these indications.

- [ ] A. The in service Flow Control Valve (AO-2-3-19) has failed closed.
- [ ] B. The Drive Water Pressure Control Valve (MO-2-3-20) is full open.
- [ ] C. A reactor scram has occurred and has not been reset.
- [x] D. The operating CRD pump shaft has sheared.

Explanation of Answer:
A. Incorrect - Charging Header Pressure would be 1400-1500 psig.
B. Incorrect - System flow would be 55-65 gpm and cooling water dp would be 20-25 psig.
C. Incorrect - System flow would be high.
D. Correct

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	

KA Information

Tier: E/APE   RO Grp: 2   SRO Grp: 2   RO Val: 3.3   SRO Val: 3.4   55.43 ✓

System:	295022	Loss of CRD Pumps
KA Group Num:	AA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	AA2.02	CRD System Status

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
CRD System Routine Insp	SO 3.8.A-2			10	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
CRD Hydraulic Syst LP	PLOT5003A	C	13	2	4

Question Data for Test: 2001 RO

88

Question: 238

Peach Bottom Unit 2 has experienced a refueling accident. Fission products were released from a dropped and damaged fuel assembly.

The automatic isolation/initiation of Reactor Building Ventilation and Standby Gas Treatment are expected due to:

- ☐ A. Equipment Cell Exhaust High Radiation.
- ☐ B. Refueling Floor Area High Radiation.
- ☑ C. Refueling Floor Vent Exhaust High Radiation.
- ☐ D. Fuel Storage Pool High Radiation.

Explanation of Answer

A. Incorrect - Equipment Cell Exhaust High Radiation is not an isolation/initiation condition.
B. Incorrect - Refuel Floor ARM does not cause isolation/initiation
C. Correct - Vent Rad High is an isolation/initiation signal and could occur with release of fission products.
D. Incorrect - Fuel Pool ARM does not cause isolation/initiation

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 3 SRO Grp: 1 RO Val: 3.3 SRO Val: 3.6 55.43 ☐

System:	295023	Refueling Accidents
KA Group Num:	AK3	Knowledge of the reasons for the following responses as they apply to:
KA Detail Num:	AK3.03	Ventilation Isolation

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Fuel Flr and Fuel Handling Prob.	ON-124 Bases	2.4.5	5	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550	B	6	7	2

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Off Normal Procedures	PLOT1550	B	6	7	3

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
ARC 218	218 A-1				

Question Data for Test: 2001 RO

89

Question: 240

Peach Bottom Unit 3 was operating at 50% power when a small steam leak occurred in the drywell.
-Drywell temperature is 250 degrees F and continuing to rise.
-Drywell pressure is 0.9 psig and steady
-The STA has reported that these conditions indicate a breach in the Primary Containment

The Technical Support Center (TSC) has recommended drywell sprays to reduce containment temperatures. Under these conditions, the Drywell Spray Logic will:

- ☐ A permit spray with the use of a bypass key.
- ☐ B permit spray until a LPCI initiation signal present.
- ☐ C prevent spray due to low torus pressure.
- ☑ D prevent spray due to low drywell pressure.

Explanation of Answer

A. Incorrect - low drywell pressure can not be bypassed
B. Incorrect - drywell spray is not permitted
C. Incorrect - system monitors drywell pressure
D. Correct

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE | RO Grp: 1 | SRO Grp: 1 | RO Val: 4.2 | SRO Val: 4.2 | 55.43 ☐

System:	295024	High Drywell Pressure
KA Group Num:	EK2	Knowledge of the interrelations between - and the following:
KA Detail Num:	EK2.11	Drywell Spray (RHR) Logic

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR Prints	M-1-S-65	D			

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
RHR	PLOT5010			1	4.s

Question Data for Test: 2001 RO

Question: 241

Peach Bottom Unit 2 has experienced a Loss of Coolant Accident (LOCA). Preparations are in progress to vent the primary containment due to a high drywell pressure per T-200-2 "Primary Containment Venting". The EO requests direction to either electrically or mechanically position the 6 inch ILRT valves.

Which one of the following identifies the method that may minimize dose to the EO and why dose would be minimized?

- [x] A Electrical positioning is from a lower dose area in Reactor Building.
- [ ] B Electrical positioning is from a remote location outside the Reactor Building.
- [ ] C Mechanical positioning is from a lower dose area in the Reactor Building.
- [ ] D Mechanical positioning is from a remote location outside the Reactor Building.

Explanation of Answer

A. Correct - Electrical positioning is performed at a panel on RB135' East wall
B. Incorrect - Electrical positioning is in RB 135' el.
C. Incorrect - Mechanical is at valve location in the same proximity as the vent line
D. Incorrect - Not possible since the valves are in the reactor building.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.3 SRO Val: 3.5 55.43 ☑

System:	295024	High Drywell Pressure
KA Group Num:	2.4	Emergency Procedures/Plan
KA Detail Num:	2.4.35	Knowledge of local Aux. Operator tasks during emergency operations including system geography and system implications.

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Containment Venting via 6" ILRT F	T-200C-2	4	3	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	C	10	8	13

Question Data for Test: 2001 RO

41

Question: 244

The bases for entering T-116 "RPV Flooding" procedure during a high drywell temperature condition is to establish:

- [ ] A flooding conditions before SRVs fail due to exceeding the SRV cabling design temperature.
- [x] B adequate core cooling due to indicated level errors from reference leg flashing.
- [ ] C flooding conditions before the level drops below the variable leg tap elevation for the Fuel Zone instruments.
- [ ] D adequate core cooling due to indicated level errors from variable leg flashing.

Explanation of Answer

A. Incorrect - Bases for 281 degrees F in accordance with T-112.
B. Correct - Ref leg official or saturation temp results in "RPV Cannot be Determined".
C. Incorrect - Minimum indication on one instrument doesn't reqire T-116 unless level cannot be determined.
D. Incorrect - The concern is for reference leg flashing

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE   RO Grp: 2   SRO Grp: 2   RO Val: 3.5   SRO Val: 3.8   55.43 ☐

System: 295028 — High Drywell Temperature

KA Group Num: EK3 — Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: EK3.02 — RPV Flooding

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102 Bases	DW/T	19	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trip Proc.	PLOT1560	C	18	8	11

Question Data for Test: 2001 RO

92

Question: 245

T-102 "Primary Containment Control" provides direction to maintain Torus level in the band of 14.5 ft. to 14.9 ft. In accordance with the TRIP Bases what is the first concern during a rising torus level transient?

- [ ] A Submerging the Reactor Building to Torus Vacuum Breaker Line.
- [x] B Excessive stress on SRV tail pipes.
- [ ] C Submergence of the Torus Spray Header.
- [ ] D Excessive stress on ECCS suction piping.

Explanation of Answer: T/L-18 Basis - Increased submergence of SRV tailpipes can cause excessive stress on SRV pipes, quenchers, and associated supports.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.4 SRO Val: 3.7 55.43

System: 295029 High Suppression Pool Water Level

KA Group Num: EK1 — Knowledge of the operational implications of the following as they apply to high suppression pool level.

KA Detail Num: EK1.01 — Containment Integrity

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met: N/A

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control - Bas	T-102 Bases	T/L-18	8	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIPS	PLOT1560			8	9

Question Data for Test: 2001 RO

93

Question: 246

For a lowering suppression pool level T-102, "Torus Level", directs that if Torus level cannot be maintained above 9.5' secure HPCI. It does not direct that RCIC be secured until < 6'.

What is the basis for securing HPCI but not RCIC at 9.5'?

- [ ] A — HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust becomes uncovered at 6'.
- [x] B — HPCI turbine exhaust becomes uncovered at 9.5', RCIC turbine exhaust is an insignificant containment input.
- [ ] C — HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust becomes uncovered at 6'.
- [ ] D — HPCI NPSH becomes a concern at 9.5', RCIC turbine exhaust is a insignificant containment input.

Explanation of Answer: Self explanatory.
RCIC is secured at 6' if it is aligned to the Torus to prevent vortexing.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE — RO Grp: 2 — SRO Grp: 1 — RO Val: 3.6 — SRO Val: 3.7 — 55.43 ☐

System: 295030 — Low Suppression Pool Water Level

KA Group Num: EK3 — Knowledge of the reasons for the following responses as they apply to low suppression pool water level.

KA Detail Num: EK3.03 — RCIC Operations

Question Source Information

Ques Source: 1999 PBAPS NRC Exam

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102	/L-11 to T/L-16	3a7-8	14	
PBAPS TRIPS	PLOT1560			8	9

Question Data for Test: 2001 RO

94

Question: 248

Peach Bottom Unit 3 has experienced a scram due to a loss of feedwater transient. With NO injection sources available, level has continued to drop.

For these conditions, what is the LOWEST reactor water level at which adequate core cooling (ACC) is still maintained.

ACC exists ONLY until reactor water level is BELOW:

- ☐ A. -172".
- ☐ B. -195".
- ☑ C. -210".
- ☐ D. -226".

Explanation of Answer:

A. Incorrect - Top of Active Fuel.
B. Incorrect - would be correct if injection was present.
C. Correct - ACC is provided by steam cooling with no RPV injection until -210".
D. Incorrect - This is the level for 2/3 core coverage.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 4.6 SRO Val: 4.7 55.43 ☐

System: 295031 — Reactor Low Water Level

KA Group Num: EK1.01 — Knowledge of the operational/implications of the following concepts as they apply to reactor low water level.

KA Detail Num: EK1.01 — Adequate Core Cooling

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Introduction to TRIPS and SAMPS	T-Bases (Intro)	5.1	18	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIPS	PLOT1560			8	8

95

Question Data for Test: 2001 RO

Question: 249

Peach Bottom Unit 2 has experienced a low reactor level transient due to a leak in the drywell. The TRIP procedures require monitoring and may direct that the use of certain level instrumentation be discontinued as indicated level approaches the lower end of the range.

This direction must be given because actual reactor water level could be:

- [ ] A — above the indicated level due to the effect of instrument reference leg flashing from high drywell temperatures.
- [ ] B — above the reference leg tap with indication downscale due to the effect of excessive water pressure on the reference leg.
- [x] C — below the variable leg tap with indicated level on scale due to the effect of area temperature on instrument calibration.
- [ ] D — below the inidicated level due to the expected instrument error experienced at the top and bottom third of scale.

Explanation of Answer

A. Incorrect - Condition of saturation in reference leg causes level to indicate > actual or upscale.
B. Incorrect - Level above reference would put indication on top of scale.
C. Correct - Off calibration temp will cause indicated level to be above actual - if level then drops indication would bottom out (max delta P) while still on scale.
D. Incorrect - Normal inaccuracies experience at top and bottom of scale are not evaluated in trip tables.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 4.4 SRO Val: 4.4 55.43 ☑

System: 295031 Reactor Low Water Level

KA Group Num: EA2 — Ability to determine and/or interpret the following as they apply to:

KA Detail Num: EA2.01 — Reactor Water Level Indication

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Trip/SAMP Curves, Tables and Lim	T-BAS	15	12-13	3	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102	Table DW/T-1		13	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS Trips	PLOT1560	C	18	8	8

Question Data for Test: 2001 RO

96

Question: 250

Peach Bottom Unit 3 has experienced a transient requiring entry into T-103 "Secondary Containment Control" on high area radiation level. A primary system breach has been verified to be discharging into the reactor building and T-103 directs a GP-4, "Manual Scram". Both temperatures and water levels remain below the Maximum Safe Operating limits.

The basis for performing a GP-4 under these conditions is to ensure:

- ☐ A the availability of Wide Range Level Instrumentation.
- ☐ B the habitability of the reactor building for personnel access.
- ☐ C compliance with the Offsite Dose Calculation Manual (ODCM).
- ☑ D a reduction in the energy being discharged into the Secondary Containment.

Explanation of Answer

A. Incorrect - Wide Range level instrumentation would be impacted by high temperatures which are still below the MSD limit.
B. Incorrect - Habitability is still not ensured due to the primary system breach.
C. Incorrect - ODCM compliance is assured by mitigation actions in T-104.
D. Correct answer.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	None

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.5 SRO Val: 3.6 55.43 ☐

System: 295033 High Secondary Containment Area Radiation Levels

KA Group Num: EK3 Knowledge of the reasons for the following responses as they apply to:

KA Detail Num: EK3.02 Reactor Scram

Question Source Information

Ques Source: New Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
T-103 Bases	T-103	SCC-9	13	11	

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Question Data for Test: 2001 RO

Question: 251

Peach Bottom Unit 3 is in MODE 3 when a complete loss of instrument air occurs. Which of the following describes the operability status of the Reactor Building/Refuel Floor Ventilation Radiation Monitors to detect a high Secondary Containment Ventilation High Radiation?

Radiation Monitors will be:

- ☐ A. Operable
- ☑ B. Inoperable
- ☐ C. Operable as long as a Reactor Building/Refuel Floor Supply Fan is still running.
- ☐ D. Inoperable until Standby Gas Treatment is started.

Explanation of Answer

A. Incorrect - No flow past detectors.
B. Correct - Loss inst air causes Reactor Bldg/Refuel Floor Exhaust and Supply dampers to close and associated fans to trip. No flow past the detectors.
C. Incorrect - Requires an associated exhaust fan to be operating.
D. Incorrect - SBGTs suction tap is upstream of detectors. No flow past detector.

Exam Level	Cognitive Level	Facility	Materials
Both	Comprehension	PBAPS	

KA Information

Tier: E/APE RO Grp: 2 SRO Grp: 2 RO Val: 3.8 SRO Val: 4.2 55.43 ☐

System:	295034	Secondary Containment Ventilation High Radiation
KA Group Num:	EA2	Ability to determine and/or interpret the following as they apply to:
KA Detail Num:	EA2.01	Ventilation Radiation Levels

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Building HVAC	PLOT5040B	II.E.6.b	28	001	6.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Loss of Instrument Air	ON-119	Attachment 1	18	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Hi-Lo Diff Pressure	ARC 317 K-5	Operator Actions	2	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Refueling Area Hi-Lo	ARC 317 L-1	Operator Actions	2	4	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Building Ventilation Syste	SO 40B.1.A	3.2	1	9	

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Question Data for Test: 2001 RO

Question: 252

Peach Bottom Unit 3 is operating at rated power, the following conditions apply:

- A high Reactor Building differential pressure exists due to a steam leak.
- Secondary Containment differential pressure is +1.25" water and rising.

Which of the following describes the INITIAL response of Secondary Containment Ventilation?

- [ ] A. Reactor Building and Refuel Floor Exhaust Fans will trip.
- [x] B. Reactor Building and Refuel Floor Supply Fans will trip.
- [ ] C. Reactor Building Equipment Cell Exhaust Fans will trip.
- [ ] D. Reactor Building and Refuel Floor Ventilation isolates.

Explanation of Answer:

A. Incorrect - Supply fans trip on high diff pressure.
B. Correct - Supply fans trip on high diff pressure.
C. Incorrect - Supply fans trip on high diff pressure.
D. Correct - An isolation does not occur

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	

KA Information

Tier: E/APE | RO Grp: 3 | SRO Grp: 2 | RO Val: 3.6 | SRO Val: 3.6 | 55.43 ☐

System: 295035 — Secondary Containment High Differential Pressure

KA Group Num: EA1 — Ability to operate and/or monitor the following as they apply to:

KA Detail Num: EA1.01 — Secondary Containment Ventilation System

Question Source Information

Ques Source: New

Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Building HVAC	PLOT5040B	II.D.1.b	19	001	4.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Hi-Lo Diff Pressure	ARC 317 K-5	Automatic Actions	1	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Refueling Area Hi-Lo	ARC 317 L-1	Automatic Actions	1	4	

Question Data for Test: 2001 RO

Question: 253

Peach Bottom Unit 3 is operating at rated power. The following conditions occur:

- A small steam leak in the drywell has raised drywell pressure to 1 psig and steady.
- A break develops in the Primary Containment.
- Reactor Building Differential Pressure is 2" water and rising slowly.

Which of the following describes the status of the Standby Gas Treatment (SBGT) System?

The Standby Gas Treatment System:

- [ ] A will be running in response to the differential pressure condition.
- [x] B can be manually started if required.
- [ ] C will be running due to the Group III Isolation Signal.
- [ ] D can not be started due to the existing differential pressure.

Explanation of Answer:

A. Incorrect - No auto start on high D/P, can be manually started.
B. Correct
C. Incorrect - No Group III Isolation will occur, Drywell pressure is affected but will lower vs. rise.
D. Incorrect - The ARC directs placing SBGT in service.

Exam Level	Cognitive Level	Facility	Materials
RO	Comprehension	PBAPS	

KA Information

Tier: E/APE   RO Grp: 3   SRO Grp: 2   RO Val: 3.6   SRO Val: 3.8   55.43 ☐

System: 295035 — Secondary Containment High Differential Pressure

KA Group Num: EK2 — Knowledge of the interrelations between - and the following:

KA Detail Num: EK2.02 — SBGT/FRVs

Question Source Information

Ques Source: New   Question Source:

Ques Mod Met:

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Secondary Containment	PLOT5009	II.B.4	9	0	1.c

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Hi-LO Diff Pressure	ARC 317 K-5	Operator Actions	2	5	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Reactor Bldg Refueling Area Hi-Lo	ARC 317 L-1	Operator Actions	2	4	

Question Data for Test: 2001 RO

Question: 257

During the execution of T-102, "Primary Containment Control", which of the following conditions would require direction be given to initiate Drywell Sprays regardless of whether Adequate Core Cooling is assured?

- ☐ A — To prevent exceeding the Pressure Suppression Pressure Limit.
- ☐ B — To maintain Drywell pressure below the Drywell Spray Initiation Limit.
- ☑ C — To mitigate the consequence of a H2 deflagration.
- ☐ D — To mitigate the consequences of containment overpressurization.

Explanation of Answer

A. Incorrect - Sprays are not used to prevent exceeding this limit.
B. Incorrect - Sprays are utilized prior to exceeding this limit but not regardless of ACC.
C. Correct - See T-102 DW/G-3.9 Bases.
D. Incorrect - If ACC is assured sprays are secured to address containment overpressurization.

Exam Level	Cognitive Level	Facility	Materials
Both	Memory	PBAPS	N/A

KA Information

Tier: E/APE RO Grp: 1 SRO Grp: 1 RO Val: 3.3 SRO Val: 3.9 55.43 ☐

System: 500000 High Containment Hydrogen Concentrations

KA Group Num: EK1 — Knowledge of the operational implications of the following concepts as they apply to high containment hydrogen concentrations.

KA Detail Num: EK1.01 — Containment Integrity

Question Source Information

Ques Source: 1999 NRC PBAPS Exam

Question Source:

Ques Mod Met: Minor wording enhancement.

References

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
Primary Containment Control	T-102 - Bases	DW/G-3.9	3.6	14	

Reference Title	Facility Ref. No.	Section	Pg #	Rev.	L.O.
PBAPS TRIPS	PLOT1560			8	9