

BRAIDWOOD
RO/SRO
EXAMINATION
JULY 17, 2002

Question Generic

Which of the following is one of the 4 Key Questions asked on the Pre-Job Briefing Checklist?

- a. How will the system respond to this manipulation?
- b. Is the intended component action correct?
- c. Is the identified action being taken on the correct Unit?
- d. What is the Worst Thing that can go wrong?

Answer d Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.1 Conduct of Operations

2.1.1 Knowledge of conduct of operations requirements. 3.7 3.8

Explanation of (A) Incorrect - part of STAR while operating. (B) Incorrect - Peer Checking (C) Incorrect - Verification
 Answer Practices (D) Correct - specifically asked on the pre-job brief checklist. The other 3 are: "What are the Critical Steps in this Task", "What are the Error Likely Situations", and "What Defenses are we relying on"

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pre-Job, Heightened..... Briefings	HU-AA-1211	Attachment 1	1	0	
Human Performance Tools & Verifications	HU-AA-101	4	3-5	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 1 RO Number: 1 SRO Number:

Question Generic

In accordance with BwAP 320-1, "Shift Staffing", the MINIMUM shift staffing requirement to comply with Tech Specs with BOTH units at power include:

RP Tech	NSO
a. 2	3
b. 2	4
c. 1	3
d. 1	4

Answer c Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.1 Conduct of Operations

2.1.4 Knowledge of shift staffing requirements.

2.3 3.4

Explanation of Answer (C) Correct - per TS 3.5.2, a RP Tech shall be onsite when fuel is in the reactor. 3 NSOs are required per 50.54

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Shift Staffing	BwAP 320-1	C	2	14	
Tech Specs	5.2.2	Organization	5.2-2	A98	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: 2001 Bwd NRC

Comment Type	Comment

Record Number: 2 RO Number: SRO Number: 1

Question Generic

In accordance with OP-AA-101-110, "Reactivity management Controls", which of the following NON-LICENSED individuals can manipulate the controls of the reactor if under the direct supervision of the licensed Reactor Operator?

- a. An individual enrolled in an approved training program
- b. A System Engineer during surveillance testing
- c. Any Non-Licensed Operator during surveillance testing
- d. Any individual directed to operate controls by the Shift Manager

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.1 Conduct of Operations

2.1.9 Ability to direct personnel activities inside the control room. 2.5 4.0

Explanation of Answer (A) Correct - per the reference, must "ensure trainees manipulating reactivity controls are enrolled in an approved training program and directly supervised by a licensed individual" (B,C,D) are then Incorrect

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactivity Management Control	OP-AA-103-104	3.5.3	2	0	
Reactivity Management Control LP	PBIG	NA	NA	0	2

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Editorially Modified

Question Source Comments: 2000 Bwd NRC 1996 Bwd NRC

Comment Type Comment

Record Number: 3 RO Number: 2 SRO Number: 2

Question Generic

The Unit 1 NSO is throttling 1AF013A, S/G 1A ISOL VLV, to adjust AFW flow to 75 gpm. In doing so, he has operated 1AF013A TWO (2) times in the last 10 seconds.

The NSO is now limited to operating the valve ____ (1) ____ times in the next 50 seconds to prevent ____ (2) ____.

____ (1) ____ ____ (2) ____

- a. 3 Overfeeding the 1A SG
- b. 3 Overheating the valve motor
- c. 4 Overfeeding the 1A SG
- d. 4 Overheating the valve motor

Answer b Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.1 Conduct of Operations

2.1.32 Ability to explain and apply all system limits and precautions. 3.4 3.8

Explanation of Answer Per BwOP AF-5, starting duties for MOV-AF013(A-H) is a max of 5 times w/l a one minute period. Prevents overheating the valve motor from excessive starting currents. (B) is only correct answer.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Normal Ops - Motor Driven AFP Startup	BwOP AF-5	E	4	16	

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment

Record Number: 4 RO Number: 3 SRO Number: 3

Question Generic

The following conditions exist on Unit 1:

- A reactor startup is in progress following an inadvertant plant trip
- The crew is performing steps of 1BwGP 100-2, "Reactor Startup"
- All control AND shutdown banks have been fully withdrawn
- The reactor is NOT critical

Which of the following describes the required operator action?

- a. Manually reinsert ALL Control and Shutdown Bank rods
- b. Emergency Borate to increase RCS boron concentration by >100 ppm
- c. Manually reinsert ONLY the Control Bank rods
- d. Immediately open the Reactor Trip Breakers

Answer a Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. 3.7 3.6

Explanation of Answer Per Attachment B, Contingency for not achieving criticality with all control rods fully withdrawn. Correct response is (A). (B) is action for criticality below Lo-2 RIL. (C) Incorrect because ALL rods must be inserted. (D) Incorrect - only applies to halted startups during severe weather conditions.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor startup procedures	1BwGP 100-2	Attach B	1	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Number(s) n

Comment Type Comment

Question Generic

Which of the following is DIFFERENT between Unit 1 and Unit 2 during current Cycle 10 operations:

- a. Shutdown Margin Limit for Modes 1,2,3 and 4
- b. DNBR - Reactor Coolant System minimum total flowrate
- c. Feedwater pressure differential pressure program
- d. Control bank insertion limits vs. % Rated Thermal Power

Answer c Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.3 (multi-unit) Knowledge of the design, procedural, and operational differences between units. 3.1 3.3

Explanation of (A) incorrect SDM both 1.3% (B) Incorrect U-1&2 =380,900 (C) Correct U-1=85-215psid, U-2=80-220psid

Answer (D) Incorrect per COLR figure 2.5.1

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
COLR	TRM COLR U-1 & U-2	COLR	5,5,17		
Power Ascension GP	BwGP 100-3	100-3A9		3,2	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments: similar to 2001 DC Cook

Comment Type	Comment

Record Number: 7 RO Number: 5 SRO Number: 5

Question Generic

Both Braidwood units undergo plant heatup and startups from cold shutdown conditons. Concerning the operation of FW009A-D, Main Feedwater Isolation Valves, they are opened earlier in the startup process on _____(1)_____ and must have startup purge logics satisfied before operating by opening bypass isolation flow control valves FW043A-D and FW046A-D on _____(2)_____.

_____ (1) _____ (2) _____

- a. Unit 1 Unit 2
- b. Unit 1 Unit 1
- c. Unit 2 Unit 2
- d. Unit 2 Unit 1

Answer a Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.4 (multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility. 2.8 3.0*

Explanation of Answer (A) Correct - opened on Unit 1 at ~200°F in GP 100-1. On Unit 2 opened at NOP/NOT in GP 100-3. Unit 2 still maintains bypass purge permissive ckts (A) is only correct answer.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Plant Heatup Procedure	1BwGP 100-1	F.39	31	17	
Power Ascension	2BwGP 100-3	F.40	44	21	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 8 RO Number: 6 SRO Number:

Question Generic

Greater than _____(1)_____ feet of water must be maintained over the top of the reactor pressure vessel flange during movement of irradiated fuel assemblies within containment in order to _____(2)_____.

_____(1)_____ _____(2)_____

- a. 23 Have sufficient water depth available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.
- b. 20 Provide sufficient water volume to allow time for the operator to recognize the indications of a dilution accident before Keff can exceed .95 delta K/K.
- c. 23 Maintain sufficient water volume as a heat sink for core cooling in the event the operating RH loop fails to provide long term decay heat removal.
- d. 20 Maintain sufficient water above the top of the fuel assemblies to ensure that the radiation levels at the operating elevation for fuel handling equipment remains below 4 mr/hr.

Answer a Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. 2.5 3.7

Explanation of Answer (A) Correct per TRM 3.9.e and TS (old) basis for minimum contained water depth during movement of fuel in containment.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
TRM		3.9.e-1	1	1	
FH LP		Ch. 52			7
TS 3/4 9.10 (old)	TS Basis (old)	Basis	9-3	A86	

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Direct From Source

Question Source Comments: 1998 Salem NRC

Comment Type Comment

Record Number: 9 RO Number: SRO Number: 6

Question Generic

The following conditions exist on Unit 1:

- Reactor was tripped from 2% power during a normal coastdown for refueling
- 7/22/02 0900 Entered Mode 3, HOT STANDBY
- 7/22/02 1300 Entered Mode 4, HOT SHUTDOWN
- 7/23/02 0600 Entered Mode 5, COLD SHUTDOWN
- 7/23/02 2300 Entered Mode 6, REFUELING

The earliest that fuel movement in the reactor vessel is allowed will be _____(1)_____ to ensure that _____(2)_____.

_____ (1) _____ (2) _____

- a. 7/25/02 1100 Short lived fission products have decayed
- b. 7/25/02 1100 Decay heat removal ability is adequate
- c. 7/26/02 1300 Short lived fission products have decayed
- d. 7/26/02 1300 Decay heat removal ability is adequate

Answer c Exam Level S Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.28 Knowledge of new and spent fuel movement procedures. 2.6 3.5

Explanation of TRM 3.9.a calls for 100 hours subcritical before fuel movements can begin. (7/22/02 @ 0900 + 100 hours (3 days, 4 hours) = 7/26/02 @ 1300. TS Basis (old) defines the basis as ensuring the short lived fission products have decayed off for radioactivity concerns

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Requirements Manual	TRM 3.9.a	9	a-1	17	
Tech Specs (old)	Basis	3/4 9.3	9-1	A56	
UFSAR		15.7			

Material Required for Examination

Question Source: Other Facility Question Modification Method: Significantly Modified

Question Source Comments: 2001 Prairie Island NRC 2000 Kewaunee NRC

Comment Type Comment

Record Number: 10 RO Number: SRO Number: 7

Question Generic

Unit 2 is being refueled following a complete core offload in accordance with the Core Loading Pattern supplied by Nuclear Fuel Services. Any deviation from the specified order of the PWR Nuclear Component Transfer List (NCTL), while transporting fuel to or from the Spent Fuel Pool or the New Fuel Storage Vault, requires the approval of ____ (1) ____ AND ____ (2) ____ before any further action is taken.

____ (1) ____

____ (2) ____

- a. System Engineering Supervisor Station Nuclear Materials Custodian
- b. Qualified Nuclear Engineer Fuel Handling Supervisor
- c. System Engineering Supervisor Fuel Handling Supervisor
- d. Qualified Nuclear Engineer Station Nuclear Materials Custodian

Answer a Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.31 Knowledge of procedures and limitations involved in initial core loading. 2.2 2.9*

Explanation of Answer Per BwAP 370-3, (A) Only correct response. (B,C,D) Incorrect - all combinations of allowable reviewers for actions that do not change the intent of the procedures.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Administrative control during refueling	BwAP 370-3	C.1.o	6	27	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 11 RO Number: SRO Number: 8

Question Generic

The following conditions exist on Unit 1:

- Reactor power is 75%, steady state, equilibrium Xenon
- All controlling systems are operating in Automatic
- Turbine Impulse pressure transmitter PT-505 fails to its 50% value.

Control rods will respond by immediately stepping ____ (1) ____ at ____ (2) ____ steps per minute.

____ (1) ____ ____ (2) ____

- a. IN 72
- b. OUT 72
- c. OUT 8
- d. IN 8

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.2 Equipment Control

2.2.33 Knowledge of control rod programming.

2.5 2.9

Explanation of (A) Correct - Tave program is 557°F-586°F, or a delta of 29°F. At 75% tave is $(.75)(29)+557=578.75^\circ\text{F}$. Tref at
 Answer 50% value is $(.5)(29)+(557)=571.5^\circ\text{F}$. A 7.25°F mismatch exists. Rods will step in at 72 steps/min with anything greater than a 5°F mismatch if Tref < Tave.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Big Notes	RD-1	Speed & Direction		3	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment

Record Number: 12 RO Number: 7 SRO Number: 9

Question Generic

Today's date is November 12, 2002 (4th quarter). You have been assigned to authorize a rad worker to perform a routine task in the Auxiliary Building for which an estimated dose of 450 mrem will be received.

Each has an exposure history this year as follows:

- A. Age 46
Cumulated TEDE dose of 600 mrem
Has a high lifetime exposure record
- B. Age 38
Cumulated TEDE dose of 48 mrem
Has 2 quarters with an absent/no dose record
- C. Age 25
Cumulated TEDE dose of 1260 mrem
Cumulated SDE dose of 6 Rem to the left hand
- D. Age 17
Cumulated TEDE dose of 80 mrem
Cumulated SDE dose of 15 mrem to the upper fore arm

Which of the above operators can be assigned the task without exceeding any of Exelon's radiation exposure limits or submitting approval for exposure limit extensions?

- a. Worker A
- b. Worker B
- c. Worker C
- d. Worker D

Answer c Exam Level S Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.3 Radiation Control

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements. 2.6 3.0

Explanation of Answer (A) Incorrect - High lifetime exposure record limits this worker to an annual dose of 1000 mrem. (B) Incorrect - allowed dose is decreased by 1250 mrem for EACH absent/no dose record on file. (C) total dose received would remain below the 2000 mrem admin limit. SDE limit of 50 R has not been exceeded (10CFR20) (D) Incorrect - minor and limited to 500 mrem for the year.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Exposure Control and Authorization	RP-AA-203	4	2-5	2	
NGET	Student Study Guide	Rad Protection			

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Topic Comment

Record Number: 13 RO Number: SRO Number: 10

Question Generic

Which of the following is an SRO responsibility?

- a. Placing the placard "Gas Decay Tank Release In Progress" on 0PM02J prior to commencing a release
- b. Performing second verification of the lineup to transfer a blowdown tank to the condensate storage tank
- c. Determining the release rate for a gas decay tank release
- d. Performing independent verification of the lineup to place a release tank on recirculation

Answer a Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.3 Radiation Control

2.3.3 Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g., waste disposal and handling systems). 1.8 2.9

Explanation of Answer A. Correct per reference. (B) Incorrect - second verifier is not required to be an SRO (C) Incorrect - Rad Protection determines release rate (D) Incorrect - IV is not required to be done by an SRO

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Waste Gas Decay Tank Release Form	BwOP GW-500T1	E.1	16	12	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: 2001 Bwd NRC

Comment Type Comment

Record Number: 14 RO Number: SRO Number: 11

Number(s) n

Question Generic

A Site Area Emergency has been declared at Braidwood Station due to a LOCA outside containment. The LOCA is into the Auxiliary Building, a direct pathway to the environment exists, and limited makeup to the RWST is available. An operator has volunteered to enter the Aux Building to locally isolate the leak. This action would significantly reduce offsite dose and has all required approvals from the TSC.

The operator has a lifetime exposure of 3200 mrem TEDE and an exposure for the current year of 230 mrem.

What is the maximum exposure this operator may receive while performing actions to isolate this leak?

- a. 5 Rem TEDE
- b. 15 Rem TEDE
- c. 25 Rem TEDE
- d. 50 Rem TEDE

Answer c Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.3 Radiation Control

2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized. 2.5 3.1

Explanation of Answer Per EP-AA-113, Personnel Protective Actions - 25 Rem TEDE is the emergency exposure limit. IT shall be voluntary and limited to once in a lifetime. (C) is correct

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Personnel Protective Actions	EP-AA-113	4.1.3	3	2	
Exposure Control and Authorization	RP-AA-203	4.1.7	4	2	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: 2001 Prairie Island NRC

Comment Type	Comment
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Record Number: 15 RO Number: 8 SRO Number: 12

Question Generic

Unit 1 is in Mode 4. Containment Purge is in progress using the Mini-purge Supply and Exhaust Fans. While the purge is in progress, 1RE-PR001, Containment Purge Effluent Rad monitor, exceeds the ALERT setpoint.

Which of the following must be performed ?

- a. MANUALLY stop the containment purge in progress
- b. VERIFY containment purge AUTOMATICALLY stops
- c. VERIFY Post LOCA Purge filter unit AUTOMATICALLY aligns
- d. MANUALLY align Post LOCA Purge filter unit

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.3 Radiation Control

2.3.9 Knowledge of the process for performing a containment purge. 2.5 3.4

Explanation of A. Correct. B. Incorrect. The AR011/12 auto isolates the purge path, not 1RE-PR001. C. Incorrect. There is
 Answer no auto alignment of the post loca purge filter unit. D. Incorrect. Procedure reference directs stopping purge
 (vice manually aligning the filter unit)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Cnmt Mini-Purge System Operation	BwOP VQ-6	E.5	2	12	
Cnmt Vent LP	I1-VP-XL-01				9

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Direct From Source

Question Source Comments: 2001 Bwd NRC

Comment Type	Comment

Record Number: 16 RO Number: 9 SRO Number: 13

Question Generic

Given the following plant conditions:

- Unit 1 is at 100% power
- Unit 2 is at 100% power
- 0PR09J "CC HX Outlet Unit 0 Radiation Monitor" is in HIGH alarm
- A confirmed High Alarm has been determined by Chemistry
- The 0 CC HX has been subsequently isolated

The crew should now verify:

- a. Only 1CC017 is closed and enter the LCO for Unit 1 CCW
- b. Only 2CC017 is closed and enter the LCO for Unit 2 CCW
- c. Both 1CC017 and 2CC017 are closed and enter the LCO for both units for CCW
- d. Both 1CC017 and 2CC017 are closed and do not need to enter a LCO for either unit

Answer c Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.3 Radiation Control

2.3.11 Ability to control radiation releases. 2.7 3.2

Explanation of for Both vent valves receive a closure signal from the common CC heat exchanger rad monitor. Must enter LCO for

Answer both units

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Specs	3.7.7	Condition A	3.7.7-1		
CC HX OUTLET UNIT O	1BwAR 1-0PR09J		1	1E1	
CC System LP	I1-CC-XL-01		7-8	0	7

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Direct From Source

Question Source Comments: 2000 Bwd NRC Exam

Comment Type Comment

Record Number: 17 RO Number: 10 SRO Number:

Question Generic

Which of the following conditions would require immediate entry into 1BwEP-0, "Reactor Trip or Safety Injection", if the condition were to occur inadvertently with the reactor operating at 100% power?

- a. Differential overcurrent on Bus 157
- b. Undervoltage on Bus 141 for >5 minutes
- c. Containment Phase A Isolation on both Trains
- d. Loss of Instrument Bus 112 with PR Instrument N-42 failed

Answer a Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. 4.0 4.3

Explanation of Answer (A) Correct - Trip of bus 157 will result in loss of 1 RCP and Rx trip when >P-8. (B) Incorrect - Loss of 1 ESF bus will not generate a Rx trip condition. (C) Incorrect - No Rx trip occurs - enter and perform OA PRI-13 "Recovery from Inadvertant Phase A" (D) Incorrect - Inst Bus 112 and N-42 are of the same channel, no Rx trip results.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Inadvertant Phase A Isolation	1BwOA PRI-13	B	1	55	
Reactor Trip or Safety Injection	1BwEP-0	B	1,2	100wog 1C	

Material Required for Examination

Question Source: Other Facility

Question Modification Method: Editorially Modified

Question Source Comments: 2001 DC Cook NRC

Comment Type	Comment

Record Number: 18 RO Number: 11 SRO Number:

Question Generic

1BwEP-3, "Steam Generator Tube Rupture", instructs the operators to maintain feedwater flow to the ruptured steam generator until narrow range level is greater than 10%.

This minimum level requirement ensures which of the following?

- a. Sufficient heat sink is available for Reactor Coolant System cooldown
- b. The ruptured steam generator tubes are covered to promote thermal stratification
- c. The ruptured steam generator does NOT become a hot-dry steam generator
- d. Radioactive steam does NOT contaminate the main steamlines

Answer b Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.6 Knowledge symptom based EOP mitigation strategies. 3.1 4.0

Explanation of EP-3 background documents - (B) Correct. Prevents ruptured SG depressurization during upcoming RCS
 Answer cooldown steps. (A) Incorrect - the ruptured SG will not be used for cooldown unless it is the only intact SG..
 (C) Incorrect - No in E-3 mitigation, with SGTR it will not become "hot and dry" (D) Incorrect - radioactive
 steam has already contaminated the steam lines.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Steam Generator Tube Rupture	BwEP-3	Step 4	8	100WO	G1C
Background Documents	EP-3		62-	1C	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments:

Comment Type	Comment
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Record Number: 19 RO Number: 12 SRO Number:

Question Generic

The following conditions exist on Unit 1:

- Bus 141 is DE-ENERGIZED
- Bus 142 is DE-ENERGIZED
- RCS pressure is 2220 psig and decreasing slowly
- Pzr level is 31% and decreasing slowly
- Preparations are being made to cool the RCS to 350°F in order to minimize further RCS inventory loss

Operators are performing steps in _____(1)_____ and are CAUTIONED NOT to decrease RCS Hot Leg temperatures below 350°F to prevent _____(2)_____:

_____ (1) _____ (2) _____

- a. 1BwCA-0.0 Loss of All AC Accumulator Nitrogen injection
- b. 1BwEP-0 Reactor Trip or SI Pressurized Thermal Shock Conditions
- c. 1BwEP-0 Reactor Trip or SI Accumulator Nitrogen injection
- d. 1BwCA-0.0 Loss of All AC Pressurized Thermal Shock Conditions

Answer a Exam Level S Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02

Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.7 Knowledge of event based EOP mitigation strategies. 3.1 3.8

Explanation of Answer (A) Correct - ESF Buses de-energized requires use of CA-0.0. During the cooldown to 350°F the operators are cautioned "To prevent injection of accumulator N2 into the RCS, hot leg temps should not be decreased to less than 350°F (B) Incorrect - EP-0 will not be in effect if cooldown steps are in progress. PTS is not a concern (C) Incorrect - EP-0 will not be in effect if cooldown steps are in progress (D) Incorrect - PTS is not a concern (per background documents) Cold leg temps will be monitored for pts

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of All AC LP	I1-CA-XL-01	II	27	4	6
Loss of All AC Power	1BwCA-0.0	step 31	39	1C	
Emergency Response Guidelines	ERG CA-0.0	step 16	118	1C	

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Significantly Modified

Question Source Comments: 2000 Bwd NRC

Comment Type Comment

Record Number: 20 RO Number: SRO Number: 14

Question Generic

An emergency call has been received by the Assist NSO in the control room, reporting a large fire in the Turbine Building.

Which of the following is the Shift Manager's responsibility while completing the checklist for a Fire/Hazmat Spill Response?

- a. Initiation of the Fire/Haz-Mat Spill Response Checklist
- b. Acquiring the location, type, size and specific area of the fire
- c. Announcing the fire location, type and size over the plant PA system
- d. Coordinating site access for the Fire Department with the Security Shift Supervisor

Answer d Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.27 Knowledge of fire in the plant procedure. 3.0 3.5

Explanation of Answer (A-C) are all part of the checklist for the Assist NSO to perform. (D) Correct - this is the Shift Managers responsibility per the checklist

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Fire / Hazmat Spill Response	BwAP 1100-16	Appendix A & D	4,12	14	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 21 RO Number: SRO Number: 15

Question Generic

A large Break LOCA concurrent with a loss of containment integrity has been INITIALLY classified as a General Emergency.

The offsite state authorities will be notified of this event on the ____ (1) ____ phone, and the NRC will be notified on the ____ (2) ____ phone.

- ____ (1) ____ ____ (2) ____
- a. Green Red
 - b. White Red
 - c. Red Green
 - d. Green White

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.29 Knowledge of the emergency plan.

2.6 4.0

Explanation of Answer (A) Dedicated lines in the MCR for NARs is GREEN and the NRC is RED.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Notifications MCR equipment	EP-AA-114				

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 22 RO Number: 13 SRO Number: 16

Question Generic

The Unit is in Mode 1 when a power supply problem in the Annunciator Cabinets results in a loss of most of the annunciators on 1PM01J, 1PM05J, and 1PM06J. Maintenance estimates 1 hour to repair.

This condition _____(1)_____ an Emergency Plan EAL threshold and requires _____(2)_____ monitoring of plant status.

_____ (1) _____ (2) _____

- a. meets Continuous
- b. meets Hourly
- c. does NOT meet Continuous
- d. does NOT meet Hourly

Answer a Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Generic Knowledge and Abilities RO Group: 1 SRO Group: 1

GENERIC

2.4 Emergency Procedures / Plan

2.4.32 Knowledge of operator response to loss of all annunciators. 3.3 3.5

Explanation of Answer (A) Correct - per Bwd EALs - MU6 and 1BwOS AN-1A AAR A.3 (B,C,D) Incorrect - the event does warrant EP classification and does require continuous monitoring

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Annunciators	1BwOS AN-1A	A.3	2	1	
Braidwood EALs	MU6	Threshold Value	5-44	6	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 23 RO Number: SRO Number: 17

Question Control Rod Drive System

The following conditions exist on Unit 1

- 90% power, steady state operating conditions
- All systems are operating normally in automatic
- Without warning, control rods begin to step
- Tave begins to increase above Tref which remains constant
- Pressurizer pressure is increasing
- Pressurizer level is increasing

These symptoms are consistent with which of the following events?

- a. One control rod has ejected from the core
- b. A SG PORV has failed open
- c. A continuous rod withdrawal is occurring
- d. A pressurizer steam space leak has developed

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

K3. Knowledge of the effect that a loss or malfunction of the Control Rod Drive System will have on the following:

K3.02 RCS 3.4* 3.5

Explanation of Answer (A) incorrect - pressurizer pressure and level would decrease. (B) incorrect - pressurizer pressure and level would decrease as Tave decreased (C) Correct - all symptoms of rod withdrawal and Tave increase (D) incorrect

- pressurizer pressure would decrease.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Rod Control LP	I1-RD-XL-01	II	1	2	1,20

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 24 **RO Number:** 14 **SRO Number:** 18

Question Control Rod Drive System

Unit 1 is at 100% reactor power, steady state

A total loss of power has occurred in data cabinet B for the Digital Rod Position Indication (DRPI) System.

What affect does this have on DRPI?

- a. System accuracy shifts to +10, -4 steps
- b. Rod at Bottom lights are LIT for all rods
- c. DRPI Urgent failure alarm annunciates
- d. Every other row of LED lights will NOT function

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

K6. Knowledge of the effect of a loss or malfunction on the following will have on the Control Rod Drive System:

K6.13 Location and operation of RPIS 3.6 3.7

Explanation of Answer (A) incorrect - accuracy for data A failure. (B) incorrect - rod at bottom lights for failure in both cabinets or dropped rod (C) incorrect - error is not in both cabinets (D) correct - system shifts to half accuracy with every other LED indicating rod position

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd System Big Notes	Dwg RD-6			2	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 25 **RO Number:** 15 **SRO Number:** 19

Question Reactor Coolant System (RCS)

Which of the following parameters should be used to differentiate between the first two (2) minutes of a moderately sized steam line break or a moderately sized RCS LOCA inside containment?

- a. Containment pressure
- b. RCS pressure
- c. Containment radiation
- d. Pressurizer level

Answer c Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 2

002 Reactor Coolant System

K3. Knowledge of the effect that a loss or malfunction of the Reactor Coolant System will have on the following:

K3.03 Containment 4.2 4.6

Explanation of Answer (C) correct - only RCS leakage will cause actual radiation levels to increase. (A,B,D) incorrect - all three will change in the same direction regardless of the transient (SLB or LOCA)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Intro to EP LP	11-EP-XL-01	acc ID chart			

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 26 RO Number: 16 SRO Number: 20

Question Reactor Coolant System (RCS)

During a recent degassing operation of the RCS, Volume Control Tank (VCT) level was increased to 70%

without any concurrent adjustment in VCT pressure as level was raised. This caused Reactor Coolant Pump (RCP) #1 seal leakoff flow to ____ (1) ____, and will require ____ (2) ____ to restore seal leakoff flows to normal.

- | | |
|---------------|-----------------|
| ____ (1) ____ | ____ (2) ____ |
| a. Increase | Venting the VCT |
| b. Increase | Opening 1CV182 |
| c. Decrease | Venting the VCT |
| d. Decrease | Opening 1CV182 |

Answer c	Exam Level B	Cognitive Level	Application	Facility: Braidwood	ExamDate: 7/17/02
Tier: Plant Systems			RO Group: 1	SRO Group: 1	

003 Reactor Coolant Pump System

A2. Ability to (a) predict the impacts of the following on the Reactor Coolant Pump System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.05 Effects of VCT pressure on RCP seal leakoff flows 2.5 2.8

Explanation of Answer (C) correct - increasing level causes pressure to increase, increasing backpressure on the RCP seals, decreasing #1 seal leakoff flow. Venting the pressure from the VCT is the required action to take while degassing the RCS. (A&B) incorrect - backpressure increases causing #1 seal leakoff to decrease. (D) incorrect - opening 1CV182 will increase seal injection flow but does not compensate for increased VCT pressure. Seal leakoffs will not return to normal parameters.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Mechanical Degassing of the RCS	BwOP CV-14	F.8-36	5-14	16	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 28 **RO Number:** 18 **SRO Number:** 22

Question Reactor Coolant Pump System (RCPS)

During normal operation, the Reactor Coolant Pump (RCP) motor windings are cooled by

_____ (1) _____
and the RCP #1 seal is cooled by _____ (2) _____.

_____ (1) _____ _____ (2) _____

- a. Air CCW
- b. Air CV
- c. CCW CV
- d. CCW CCW

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

003 Reactor Coolant Pump System

K4. Knowledge of Reactor Coolant Pump System design feature(s) and or interlock(s) which provide for the following:

K4.04 Adequate cooling of RCP motor and seals 2.8 3.1

Explanation of Answer (B) Correct - air cools the motor windings and water (CV) cools the #1 seal (seal injection)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCP Lesson Plan	11-RC-XL-01	II	2,11,49	1	3,4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 29 **RO Number:** 19 **SRO Number:** 23

Question Chemical and Volume Control System (CVCS)

The following plant conditions exist on Unit 1

- A small break LOCA has occurred
- Control Room indications suggest that the Pressurizer is solid
- RVLIS head and plenum indicate 100%

Which of the following describes the effect of changes in charging and letdown on the Reactor Coolant System under these conditions?

- a. Small mismatches between charging and letdown may cause large and sudden RCS pressure changes.
- b. Large mismatches between charging and letdown may be required to induce small changes in RCS pressure.
- c. Small mismatches between charging and letdown may cause large and sudden pressurizer level changes.
- d. RCS pressure will respond exactly the way it responds to charging and letdown changes with a bubble present in the Pzr.

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Chemical and Volume Control System controls including:

A1.09 RCS pressure and temperature 3.6 3.8

Explanation of Answer (A) Correct - water is an incompressible fluid. In a solid condition, small variations in inventory (charging / letdown flows) will result in large variations in pressure.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CVCS LP	I1-CV-XL-01				
RCS Fill and Vent	BwOP RC-3	F	16-19	14	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments:

Comment Type **Comment**

Record Number: 30 **RO Number:** 20 **SRO Number:** 24

Question Chemical and Volume Control System (CVCS)

Two (2) minutes following a reactor trip with a loss of off-site power, which of the following motor operated valves will NOT have power available? (Assume no operator actions)

- a. 1CV8104 "Emergency Boration Valve"
- b. 1CV112D "Charging Pump Suction from RWST Valve"
- c. 1CV8105 "Charging Pump to Reactor Coolant Sys Isolation Valve"
- d. 1CV8109 "Positive Displacement Pump Recirc Valve"

Answer d **Exam Level** R **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System

K2. Knowledge of bus power supplies to the following:

K2.05 MOVs 2.7 2.9

Explanation of Answer (D) correct - the only non-esf powered MOV in the list. 1CV8109 is powered from 133V1. (A,B,C) incorrect - these are ESF powered valves and will be energized following a loss of off-site by the respective EDGs

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CV Electrical Lineup	BwOP CV-E1		1,2	5	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 31 **RO Number:** 21 **SRO Number:**

Question Residual Heat Removal System (RHRS)

The following conditions exist on Unit 1:

- RCS temperature is 340°F
- RCS pressure is 345 psig
- Pzr level is 54%
- 1A RH train is being aligned for shutdown cooling
- 1B RH train is aligned for injection

Shortly after placing the 1A RH train in the shutdown cooling mode, the NSO notices RCS wide range pressure and Pzr level decreasing. A Field Operator on rounds informs the NSO that the 1B RH pump suction relief valve sounds like it is lifting.

Which of the following describes the possible cause of this event? (Evaluate each response separately)

- a. 1SI8809B, RH TRN to RCS Cold Leg Injection valve, was inadvertently closed
- b. 1CV131, Letdown Pressure Control valve, was left in AUTO when the 1A RH Pump was started
- c. 1RH8716A, RH Disch. Header X-Tie valve, was left open prior to starting the 1A RH Pump
- d. 1RH8701A, RC Loop A to RH Pump Suction valve, inadvertently closed due to an instrument failure

Answer C **Exam Level** B **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

Residual Heat Removal System

K3. Knowledge of the effect that a loss or malfunction of the Residual Heat Removal System will have on the following:
 K3.01 RCS 3.9 4.0

Explanation of Answer (A) Incorrect - closure of the discharge will not increase suction pressure (B) Incorrect - RCS is not solid (C) Correct - per reference, closure of the disch X-tie valve is performed to prevent overpressurization of the idle pump suction header. (D) Incorrect - closure of the RC Suction valve will not cause dish press to rise on the opposite train

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Placing the RH system in S/D Cooling	BwOP RH-6	F	12	26	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:

Comment Type **Comment**

Record Number: 32 **RO Number:** 22 **SRO Number:** 25

Question Residual Heat Removal System (RHRS)

The following conditions exist on Unit 1

- Mode 5 operations with Train A RHR aligned and providing shutdown cooling
- RCS temperature is 190°F
- RCS pressure is 330 psig
- Solid plant ops
- RH letdown in service
- 1PCV-131 controlling RCS pressure in automatic

Which of the following describes the INITIAL primary system response if the operating RHR Pump trips?

RCS Temperature	RCS Pressure
-----------------	--------------

- | | |
|-------------|----------|
| a. Increase | Decrease |
| b. Increase | Increase |
| c. Decrease | Decrease |
| d. Decrease | Increase |

Answer b **Exam Level** B **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

Residual Heat Removal System

K5. Knowledge of the operational implications of the following concepts as they apply to the Residual Heat Removal System:

K5.05 Plant response during "solid plant": pressure change due to the relative incompressibility of water 2.7* 3.1*

Explanation of Answer (B) Correct. Loss of RH cooling flow will allow RCS to heat up. 1CV-131 will sense pressure drop as RH pump is tripped and close to raise RCS pressure (A) incorrect - pressure will increase. (C&D) Incorrect - RCS will heat up as RH cooling is lost

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
General Operating Procedure	1BwGP 100-1	D.3.c	5	17	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 33 **RO Number:** 23 **SRO Number:** 26

Question Emergency Core Cooling System (ECCS)

The following conditions exist on Unit 2

- A large break LOCA is in progress
- RWST level decrease requires the operators to transfer to Cold Leg Recirculation
- 2RH8702A and B, "RC Loop 2C to RH Pump 2B Suction Isolation Valves", are both CLOSED
- 2SI8811B, "Containment Sump 2B Isolation Valve", is OPEN

Which of the following actions MUST be performed to OPEN 2SI8804B, "2B RH Hx to CV/SI Pump Suction Isolation Valve"?

- a. CLOSE 2SI8813 "SI Pump Common Miniflow Isolation Valve"
- b. OPEN 2SI8807B "SI/CV Pumps Suction Header Crosstie Valve"
- c. OPEN 2CS009B "CS Pump 2B Sump Suction Valve"
- d. CLOSE 2SI8812B "RH Pump 2B Suction from RWST Isolation Valve"

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 2

006 Emergency Core Cooling System

K4. Knowledge of Emergency Core Cooling System design feature(s) and or interlock(s) which provide for the following:
 K4.17 Safety Injection valve interlocks 3.8 4.1

Explanation of Answer (A) Correct - closing either (2SI8920 AND 2SI8814) or closing 2SI8813 satisfies the rest of the interlock to open the RH crosstie 2RH8804B. (B,C,D) Incorrect - none are in the interlock circuitry

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
GPs - Main Control Board Valve Interlocks	2BwGP 100-1A3		2	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 34 RO Number: 24 SRO Number: 27

Question Pressurizer Relief Tank/Quench Tank System (PRTS)

The NSO has noted an increasing level in the Pressurizer Relief Tank (PRT).

Which one of the following RELIEF VALVES might be discharging to the PRT?

- a. 1CV8118, Charging Pump discharge relief valve
- b. 1CV8117, Letdown line orifice relief valve
- c. 1CC9426A-D, RCP thermal barrier relief valves
- d. 1SI8856A/B, RH Pump discharge relief valves

Answer b **Exam Level** R **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

007 Pressurizer Relief Tank/Quench Tank System

A3. Ability to monitor automatic operations of the Pressurizer Relief Tank/Quench Tank System including:

A3.01 Components which discharge to the PRT 2.7* 2.9

Explanation of Answer (A) Incorrect - 1CV8118 relieves to the VCT (B) Correct - relieves to the PRT (C) Incorrect - relieves to the Cnmt Bldg Floor Drain Sump (D) Incorrect - relieves to the HUT

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
P&ID	M-64 sheet 5			BE	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: 2001 Bwd NRC

Comment Type	Comment
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Record Number: 35 **RO Number:** 25 **SRO Number:**

Question Component Cooling Water System (CCWS)

A leak in which of the following components will result in an automatic closure of 1CC017, "Component Cooling Surge Tank Vent Valve"

- a. Seal Water Heat Exchanger
- b. Spent Fuel Pool Heat Exchanger
- c. Letdown Heat Exchanger
- d. Waste Gas Compressor Heat Exchanger

Answer C **Exam Level** R **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

008 Component Cooling Water System

K1. Knowledge of the physical connections and/or cause-effect relationships between Component Cooling Water System and the following:

K1.03 PRMS 2.8* 3.0

Explanation of Answer (C) correct - RCS letdown is at a higher pressure than the CCW system and will result in inleakage to CCW. (A,B,D) are all at a lower operating pressure than CCW and will result in CC outleakage. (see 1BWOA PRI-6)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Big Notes	CC-2	Leakage Sourcres		4	
Component Cooling Malfunction	1BWOA PRI-6	Attach B	27-30	100	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments:

Comment Type	Comment

Record Number: 36 **RO Number:** 26 **SRO Number:**

Question Pressurizer Level Control System (PZR LCS)

Which of the following Pressurizer level channels is COLD Calibrated, causing it to indicate LOWER than actual level at normal operating temperature and pressure?

- a. LT-459
- b. LT-460
- c. LT-461
- d. LT-462

Answer d **Exam Level** R **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

011 Pressurizer Level Control System

K4. Knowledge of Pressurizer Level Control System design feature(s) and or interlock(s) which provide for the following:

K4.03 Density compensation of PZR level 2.6 2.9

Explanation of Answer (D) Correct - LT-462 is NOT density compensated for NOP/NOT but is calibrated to read accurately at cold conditons.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Pressurizer LP	I1-RY-XL-01	II	15	2	18
Bwd Curve Book	BwCB-1 fig 31	figures	1of1	0	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment

Record Number: 37 **RO Number:** 27 **SRO Number:**

Question Reactor Protection System

Which of the following reactor protection system trips serves as a BACK-UP to the Power Range Neutron

Flux - High trip and is designed to ensure that the allowable heat generation rate (kw/ft) of the fuel is NOT exceeded?

- a. OTdT
- b. OPdT
- c. Pzr low pressure
- d. RCS low flow

Answer b Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 2

012 Reactor Protection System

K5. Knowledge of the operational implications of the following concepts as they apply to the Reactor Protection System:
 K5.02 Power density 3.1* 3.3

Explanation of Answer Per TS 3.3.1 - basis (B) is only correct response

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Specs	Basis	B 3.3.1	17	24	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 40 RO Number: 30 SRO Number: 29

Question Engineered Safety Features Actuation System (ESFAS)

The following conditions exist on Unit 1:

- A RCS LOCA has occurred.
- Safety Injection and all ESFAS equipment has actuated and is functioning as designed.
- The Emergency Director has declared an ALERT condition exists
- The crew is performing the actions of 1BwEP ES-1.2, "Post LOCA Cooldown and Depressurization"
- No CSF higher than YELLOW is in effect
- Annunciator 1-6-B7," RWST LEVEL LO-2" has just alarmed

Which of the following actions should be taken?

- a. A Site Evacuation should be ordered AND the people directed to assemble at the New Training Building
- b. A plant announcement should be made warning personnel to restrict entry into the Aux Building due to potential high radiation
- c. The event should be reclassified as a General Emergency AND the NRC, State and local governments notified immediately
- d. Protective Action Recommendations (PARs) determined AND State and local governments notified within 1 hour following evaluation

Answer b **Exam Level** S **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
 013 Engineered Safety Features Actuation System
 2.1 Conduct Of Operations
 2.1.14 Knowledge of system status criteria which require the notification of plant personnel. 2.5 3.3
Explanation of Answer (A) Incorrect - No radiological safety hazzard warrenting evacuation is in progress. (B) Correct - Switchover to recirc may cause high rad levels in the aux building. (C) Incorrect - No General Emergency conditions have been met. (D) Incorrect - PARs are only required for a General Emergency

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Transfer to cold leg recirc	1BwEP ES-1.2	CAUTION	10	100	
Generating Stations Emergency Plan	GSEP - Braidwood Annex	Initiating Conditions	3,4	6	
Emerg Classifications & PARs	EP-AA-111		1-16		

Material Required for Examination

Question Source: Other Facility **Question Modification Method:** Editorially Modified

Question Source Comments: 2001 Prairie Island NRC

Comment Type **Comment**

Record Number: 41 **RO Number:** **SRO Number:** 30

Question Engineered Safety Features Actuation System (ESFAS)

Unit 2 is presently at 90% power and shutting down due to an extended loss of Instrument Bus 214. All systems are in automatic when a loss of reactor coolant occurs. When pressurizer pressure reaches 1829

psig on 2/4 channels, which of the following will occur?

- a. Automatic SI will occur. Train A ECCS equipment will automatically start. Train B ECCS equipment must be manually started.
- b. Automatic SI will NOT occur. Train A and Train B ECCS equipment will automatically start when SI is manually actuated.
- c. Automatic SI will occur. Train A and Train B ECCS equipment will automatically start when SI is automatically actuated.
- d. Automatic SI will NOT occur. Train A ECCS equipment must be manually started. Train B ECCS equipment cannot be started.

Answer a **Exam Level** R **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System

K2. Knowledge of bus power supplies to the following:

K2.01 ESFAS/safeguards equipment control 3.6* 3.8

Explanation of SI will automatically actuate, both trains. Train B ESF Loads however have lost the relay (energize to actuate)

Answer and will not auto start as designed. They can be manually started. (A) Correct (B) incorrect - SI will actuate, Train B will not auto start (C) Train B will not auto start (D) Incorrect - Train A will auto start

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Instrument Bus	1BWOA ELEC-2	table D	18	100	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 42 **RO Number:** 31 **SRO Number:**

Question engineered Safety Features Actuation System (ESFAS)

Concerning the Engineered Safety Features Actuation System (ESFAS), there are ____ (1) ____ channels of narrow range steam generator level instrumentation on each steam generator which input to ____ (2) ____ independent safety trains of ESF.

- | | |
|---------------|---------------|
| ____ (1) ____ | ____ (2) ____ |
| a. 2 | 4 |
| b. 2 | 2 |
| c. 4 | 2 |
| d. 4 | 4 |

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1

013 Engineered Safety Features Actuation System

K5. Knowledge of the operational implications of the following concepts as they apply to the Engineered Safety Features Actuation System:

K5.01 Definitions of safety train and ESF channel 2.8 3.2

Explanation of channels There are 2 independent trains of ESF (A&B). Each train will receive an input from each one of 4 level channels

Answer on each steam generator for ESF purposes. (TS 3.3.2) (C) is only correct combination

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ESF LP	I1-EF-XL-01	II		2	5,7
Tech Specs	LCO	3.3.2	13	Am115	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 43 RO Number: 32 SRO Number: 31

The following conditions exist on Unit 1:

- Reactor power is holding steady at 1×10^{-8} amps during a normal reactor startup
- Individual and group position indicators show all control bank D rods at 120 steps withdrawn

When the NSO begins to withdraw control rods to raise reactor power, the IR NIS indication suddenly drops by 1/3 decade and continues to decrease at a negative (-).25 DPM. There is no significant change in RCS Tave. The control bank D step counters now read 121 steps for both D1 and D2 groups. DRPI indicators for rods D-12, M-4, and H-8 indicate 0 steps. All other rod position indicators (DRPI) are unchanged.

Which of the following has occurred based on these indications?

- a. The control bank step counters and associated DRPI indicators, along with the NIS indications are consistent with multiple dropped rods.
- b. The individual rod position indicators appear to have failed, more than a single dropped rod would have resulted in a reactor trip
- c. The control bank D group 2 step counter has failed, it should also read 0 steps if the rods in this group are fully inserted
- d. Either the control bank D group step counter or 3 DRPI indicators have failed, not enough information is provided to determine which

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 1
 014 Rod Position Indication System
 K1. Knowledge of the physical connections and/or cause-effect relationships between Rod Position Indication System and the following:
 K1.02 NIS 3.0 3.3

Explanation of Answer Indications provided are consistent with multiple dropped rods. (A) is correct. (B) incorrect - the reactor does not trip from neg rate (C) incorrect - group counters are demand indicators only. (D) incorrect - given NIS response, DRPI or group demand counters have no independent affect on NIS

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Ops - Dropped or Misaligned Rod	1BwOA ROD-3	Symptoms	1	101	
Bwd Big Notes	RD-6	DRPI Indicatons	1	2	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type Comment

Topic

Question Nuclear Instrumentation System

Which of the following describes the effects of a short unintentional emergency boration on the reactor at 75% power. Assume that control rods are in MANUAL.

- a. Tave initially decreases causing reactor power to decrease. Tave then increases to approximately the initial value.
- b. Tave initially increases causing reactor power to decrease. Tave then decreases to approximately the initial value.
- c. Reactor power initially decreases causing Tave to decrease. Reactor power then increases to approximately the initial value.
- d. Reactor power initially decreases causing Tave to increase. Reactor power then increases to approximately the initial value.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

A1. Ability to predict and/or monitor changes in parameters associated with operating the Nuclear Instrumentation System controls including:

A1.07 Changes in boron concentration 3.3* 3.4

Explanation of Answer Boration is an added poison, fewer neutrons available for absorption in the fuel, reactor power decreases. The decrease in reactor power results in a lower Tave. As Tave decreases, (+) reactivity is added causing reactor power to increase. (C) correct response.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Normal Operating Procedures	BwOP CV-6	Precautions	2	14	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments:

Comment Type	Comment

Number(s) n

Record Number: 45 **RO Number:** 34 **SRO Number:** 33

Question Nuclear Instrumentation System

The following conditions exist on Unit 2:

- A normal reactor startup is in progress
- Reactor power is steady at 1000 cps
- PR NI channel N-41 has failed low
- Operators have completed performing 2BwOA INST-1, "Nuclear Instrumentation Malfunction" for the failed PR channel

Seconds later the control power fuses on PR channel N-43 both indicate blown

Which of the following describes the next action to take:

- a. Verify all rod at bottom lights LIT
- b. Manually reinsert all control bank rods
- c. Manually reinsert all control and shutdown banks
- d. Complete the startup to <P-10 using only IR instruments

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1
 015 Nuclear Instrumentation System
 K6. Knowledge of the effect of a loss or malfunction on the following will have on the Nuclear Instrumentation System:
 K6.04 Bistables and logic circuits 3.1 3.2

Explanation of automatic trip on PR instruments is active, even at < indicted PR power. SR instruments are de-energized due to P-10 pickup. (A) is correct. Candidates may confuse 2 PR channel failures with P-10 actuation and blocking of SR instruments which make distractors (B,C,D) plausible but incorrect

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Braidwood Big Notes		NI-2	1	4	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 46 RO Number: 35 SRO Number: 34

Question Non-Nuclear Instrumentation System (NNIS)

Which of the following Main Control Board recorders provides the operator with the option of SELECTING specific channels for trending?

- a. 1LR-930 RWST Level
- b. 1PR-937 Containment Pressure
- c. 1PR-0514 Steam Generator Pressure
- d. 1FR-0510 Steam Generator Steam Flow/Feed Flow

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems RO Group: 2 SRO Group: 2

016 Non-Nuclear Instrumentation System

A4. Ability to manually operate and/or monitor in the control room:

A4.02 Recorders 2.7 2.6*

Explanation of Answer (A,B,C) have no capability to select optional channels on the MCB. (D) has 2 channels available with a MCB selector switch on 1PM04J

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
MCB Layout SGWLC LP	11-FW-XL-01	II	11-14	2	S.FW2-10

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 47 RO Number: 36 SRO Number: 35

Question In-Core Temperature Monitor (ITM) System

A Loss of Coolant Accident has occurred, core exit thermocouple (CETC) temperatures are reading 690°F and increasing.

Which of the following describes the expected response of the CETCs as the Reactor Coolant System and core exit temperatures continue to increase. (Assume no core cooling is present)

The CETC's will indicate ..

- a. lower than actual temperature above 700°F, and will stop indicating altogether as temperatures exceed 1200°F.
- b. accurately up to 1800°F, and can be used for trending purposes up to 2300°F.
- c. higher than actual temperature above 700°F, and cannot be relied upon for accurate indication above 1200°F.
- d. accurately up to 1200°F, and will fail completely above 1800°F.

Answer b Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1

017 In-Core Temperature Monitor System

K1. Knowledge of the physical connections and/or cause-effect relationships between In-Core Temperature Monitor System and the following:

K1.02 RCS 3.3 3.5

Explanation of Answer CETCs usable range is 200-1800°F. They are expected to indicate up to 2300°F with reduced accuracy. (B) is correct.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Inadequate Core Cooling System LP	I1-IT-XL-01	II	8	1	5

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 49 RO Number: 38 SRO Number:

Question Containment Cooling System (CCS)

The following conditions exist on Unit 2:

- The reactor is at 100% power, steady state
- 2A and 2C RCFCs are running in high speed
- 2B and 2D RCFCs are in standby
- A reactor trip and Safety Injection occur due to a large break RCS LOCA.

Which of the following describes the low speed start response of the RCFCs after the SI signal is received?

- a. All four RCFCs start 20 seconds after the SI signal
- b. 2A and 2C RCFCs start 20 seconds after the SI signal, 2B and 2D RCFCs start immediately
- c. All four RCFCs start immediately after the SI signal
- d. 2A and 2C RCFCs start immediately after the SI signal, 2B and 2D RCFCs start 20 seconds later

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

022 Containment Cooling System

A4. Ability to manually operate and/or monitor in the control room:

A4.01 CCS fans 3.6 3.6

Explanation of Answer Fans already running in slow speed will remain running - none are in this example. All high speed breakers trip (2A & 2C). After a 20 second time delay, slow speed breakers close on non-running fans. (A) correct response.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Big Notes	Containment Cooling	VP-3		5	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment

Record Number: 50 **RO Number:** 39 **SRO Number:** 36

Question Containment Cooling System (CCS)

The following conditions exist on Unit 1:

- A small RCS LOCA has occurred
- The crew is performing 1BwOA PRI-1, "Excessive Primary Plant Leakage"
- Containment pressure is 2.9 psig and increasing slowly
- RCFC 1C low speed breaker auto SI closure relay failed and was declared inoperable
- Annunciator 1-3-E5, "RCFC LOCAL CONT", is LIT.
- RCFC 1C is in LOCAL control at the RSDP and running in HIGH SPEED

Which of the following describes the available operation of the 1C RCFC in the current configuration if containment pressure continues to rise to 4.0 psig?

- a. The high speed breaker will automatically trip, the low speed breaker CAN ONLY be closed from the RSDP 20 seconds later.
- b. The high speed breaker will NOT automatically trip, the low speed breaker CAN NOT be closed at either the RSDP or 1PM06J.
- c. The high speed breaker will automatically trip, the low speed breaker CAN BE manually closed from 1PM06J 20 seconds later.
- d. The high speed breaker will NOT automatically trip, the low speed breaker MUST BE closed locally after the high speed breaker is tripped.

Answer c Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1
 022 Containment Cooling System
 2.4 Emergency Procedures / Plan
 2.4.31 Knowledge of annunciators alarms and indications, and use of the response instructions. 3.3 3.4

Explanation of RCFC LOCAL CONT indicates that the high speed breaker for the RCFC is in local control. Low speed breaker

Answer operation is unaffected by the Local/remote switch. (C) is correct (A) Incorrect - low speed operation is not available at the RSDP. (B) Incorrect - high speed breaker will trip on the SI, 1PM06J low speed operation is available. (D) Incorrect - High speed breaker will trip, 1PM06J low speed operation is available.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Annunciator Response procedures	BwAR 1-3-E5	NOTE	1	5	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 51 RO Number: 40 SRO Number:

Question Containment Spray System (CSS)

Which of the following valves is interlocked with OPENING of the 1A CS Pump Sump Suction Isolation Valve, 1CS009A?

- a. 1SI8809A RHR Cold Leg Injection Isolation Valve
- b. 1SI8811A 1A Containment Sump Isolation Valve
- c. 1CS007A 1A CS Pump Discharge Isolation Valve
- d. 1CS001A 1A CS Pump RWST Suction Isolation Valve

Answer b Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 1

026 Containment Spray System

K1. Knowledge of the physical connections and/or cause-effect relationships between Containment Spray System and the following:

K1.01 ECCS 4.2 4.2

Explanation of Answer Interlocks for opening 1CS009A are: 1SI8811A Open, 1RH8701A Closed, CS to Open. (B) is only correct answer. (A,C,D) valves are not in interlock circuitry and are incorrect

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CS LP	11-XL-CS-01	II	16	2	S.CS1-08-A
Bwd Big Notes	Containment Spray	CS-1		5	

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 52 RO Number: 41 SRO Number: 37

Question Containment Spray System (CCS)

The following conditions exist on Unit 1:

- A reactor trip and safety injection have occurred due to a large break RCS LOCA
- All ECCS equipment functioned normally upon receipt of the SI signal
- Five (5) minutes after the SI was received, Containment Spray (CS) actuation setpoint was exceeded
- Train B of CS started as designed
- Train A of CS did NOT auto start and could NOT be manually started from the Control Room

Which of the following annunciators, if received JUST PRIOR to reaching the CS actuation setpoint, would result in the above status for CS?

- a. 1-21-A4 BUS 133X/133Y FD BRKR TRIP
- b. 1-22-A4 BUS 134X/134Y FD BRKR TRIP
- c. 1-21-A10 BUS 131X FD BRKR TRIP
- d. 1-22-A10 BUS 132X FD BRKR TRIP

Answer c Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 1

026 Containment Spray System

K2. Knowledge of bus power supplies to the following:

K2.02 MOVs 2.7* 2.9

Explanation of Interlocks - to auto start 1CS019 must auto open. Power supply is 131X1. For CS to be manually started, Answer 1CS007A must be closed. Power supply is 131X5. To start in recirc, 1SI8811A must be open but we are not in recirc yet. (C) is only correct answer. (D) incorrect - train B PS. (A&B) incorrect - non esf power supplies.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
CS lesson Plan	11-XL-CS-01	II	12-15	2	S.CS1-16
Bwd Big Notes	Containment Spray	CS-1		5	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 53 RO Number: 42 SRO Number: 38

Question Containment Iodine Removal System

Containment Iodine removal via the Containment Spray pumps requires power from ____ (1) ____ busses and Containment Charcoal Filter Fans which require power from ____ (2) ____ busses.

____ (1) ____ ____ (2) ____

- a. ESF ESF
- b. Non-ESF ESF
- c. ESF Non-ESF
- d. Non-ESF Non-ESF

Answer c Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 3 SRO Group: 2

027 Containment Iodine Removal System

K2. Knowledge of bus power supplies to the following:

K2.01 Fans 3.1* 3.4*

Explanation of CS Pumps are powered via ESF busses 141 and 142. Cnmt Charcoal Filter Fans are powered via Non-ESF busses 133 and 134 (C) Correct

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Electrical Lineups	BwOP CS-E1		1	2	
Electrical Lineups	BwOP VP-3		1	0E1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 54 RO Number: 43 SRO Number:

Question Hydrogen Recombiner and Purge Control System (HRPS)

Under normal plant conditions, opening the Post LOCA Hydrogen Monitoring outside suction and discharge isolation valves, 1PS228A/B - 1PS230A/B, is accomplished in the Main Control Room from _____(1)_____, and indication of containment hydrogen concentration will normally be displayed on the _____(2)_____ scale.

- | | |
|--|-----------------|
| _____ (1) _____ | _____ (2) _____ |
| a. 1PM11J Unit 1 Containment Isolation Panel | HI |
| b. 0PM02J Unit Common Ventillation Panel | LO |
| c. 0PM02J Unit Common Ventillation Panel | HI |
| d. 1PM11J Unit 1 Containment Isolation Panel | LO |

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 3 SRO Group: 2

028 Hydrogen Recombiner and Purge Control System

A4. Ability to manually operate and/or monitor in the control room:

A4.03 Location and operation of hydrogen sampling and analysis of containment atmosphere, including alarms and indications 3.1 3.3

Explanation of Answer (D) Correct - Controls and indications for 1PS228, 229, and 230A&B valves are all located on the Containment Isolation Panel, 1PM11J. PS343 and PS344 are normally selected to indicate on the LO range (B&C) Incorrect - controls are not located on 0PM02J (A) Incorrect - normally selected to LO range

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Normal Operating Procdures	BwOP PS-9	F.1	2	8E2	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 55 RO Number: 44 SRO Number: 39

Question Containment Purge System (CPS)

The following conditions exist on Unit 1:

- Reactor is at 100%, steady state power
- Containment purge is in progress and is being performed under BwRP 6110-13T1, "Containment Release Form"
- During the release, Health Physics requested 1RE-PR001 be placed in purge for a filter change

Which of the following actions must be taken prior to placing 1RE-PR001 in purge to comply with the requirements of BwRP 6110-13T1 and RETS 2.2-1a?

- a. Suspend the containment release of radioactive effluents via this pathway
- b. Obtain continuous samples of this pathway with auxiliary sampling equipment
- c. Restore the monitor to operable status before the next 30 hour sample is required
- d. Verify that Cnmt rad monitor 1RE-AR011 is operable and continue the release

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

029 Containment Purge System

A2. Ability to (a) predict the impacts of the following on the Containment Purge System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.04 Health physics sampling of containment atmosphere 2.5* 3.2*

Explanation of Answer isolating 1RE-PR001 renders the noble gas activity monitor inoperable. RETS 2.2.1a requires immediate suspension of purging via this pathway if this occurs. (A) Correct. (B) incorrect - this is an allowable option only for the iodine and particulates functions of 1RE-PR001. (C) incorrect - continuous samples are still required. (D) incorrect - substituting AR011 is not allowed.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
ODCM RETS - Gaseous effluent monitoring	RETS 2.2-1a		18-20	5	
Containment Release Form	BwRP 6110-13T1	B	8	7	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments: similar to recent Bwd LER event

Comment Type **Comment**

Record Number: 56 **RO Number:** 45 **SRO Number:** 40

Question Containment Purge System (CPS)

A reactor trip and safety injection has occurred on Unit 1. When performing 1BwEP-0, "Reactor Trip or Safety Injection", some of the Group 6 CNMT Vent Isol monitor lights are NOT LIT.

What are the MINIMUM actions that must be taken to comply with this step of 1BwEP-0?

- a. STOP any running VQ fans
- b. OPEN VQ isolation valve(s) as necessary
- c. STOP any running VQ fans AND manually CLOSE VQ isolation valve(s) as necessary
- d. START all non-running VQ fans AND OPEN VQ isolation valve(s) as necessary

Answer c Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems

RO Group: 2 SRO Group: 2

029 Containment Purge System

2.4 Emergency Procedures / Plan

2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. 4.0 4.0

Explanation of Answer (A) Incorrect - VQ isol valves must also be closed (B) Incorrect - running VQ fans must also be closed AND valves closed (C) Correct - per 1BwEP-0, RNO for step 8 is numbered (1,2) requiring fans to be stopped AND valves closed. (D) Incorrect - Fans must be stopped and valves closed to energize the group 6 monitor lights (and per procedure - see C)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Big Notes - Cnmt Purge	VP-2		1	5	
Reactor Trip or SI	1BwEP-0	step 8 and RNO	7	100	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 57 RO Number: 46 SRO Number:

Question Spent Fuel Pool Cooling System (SFPCS)

If a leak develops on the discharge of the Spent Fuel Pool Heat Exchanger while the cooling loop is in operation, the Spent Fuel Pool will lose a MAXIMUM of _____(1)_____ before the FC Pump loses suction. Using BwOP FC-11, makeup water will be added back to the Spent Fuel Pool via the _____(2)_____.

- | | |
|-----------------|-------------------------------------|
| _____ (1) _____ | _____ (2) _____ |
| a. 4 feet | Refueling Water Storage Tank (RWST) |
| b. 4 inches | Volume Control Tank (VCT) |
| c. 4 inches | Refueling Water Storage Tank (RWST) |
| d. 4 feet | Volume Control Tank (VCT) |

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

033 Spent Fuel Pool Cooling System

A2. Ability to (a) predict the impacts of the following on the Spent Fuel Pool Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.03 Abnormal spent fuel pool water level or loss of water level 3.1 3.5

Explanation of Answer (A) Correct - the pump suction has stops 4 feet below normal water level. The RWST is an available source of makeup water. (B) Incorrect - the 4 inches relates to the anti-siphon hole on the SFP cooling discharge to the pool. The VCT also is not an available option for makeup water. (C) incorrect - the 4 inches relates to the anti-siphon hole on the SFP cooling discharge to the pool (D) Incorrect - the VCT is not an available source of makeup water.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Spent Fuel Pool Level Adjustment Proc	BwOP FC-11	F	3	20	
Spent Fuel Pool Cooling Lesson Plan	I1-FC-XL-01	fig 51-13		1	5,6

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 58 **RO Number:** 47 **SRO Number:** 41

Question Fuel Handling Equipment System (SHES)

What prevents raising an irradiated fuel assembly out of the Spent Fuel Pool using the new fuel elevator?

- a. Controls for the new fuel elevator will only travel in one direction - there is no upward motion available
- b. Upward motion of the new fuel elevator is stopped if surface radiation levels approach 100 mr/hr
- c. An upward motion interlock prevents lifting any loads greater than 1200 lbs with the new fuel elevator
- d. A slack cable interlock prevents raising the much lighter spent fuel assembly via the new fuel elevator

Answer c Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 3 SRO Group: 2

034 Fuel Handling Equipment System

A3. Ability to monitor automatic operations of the Fuel Handling Equipment System including:

A3.02 Load limits 2.5* 3.1

Explanation of Answer (C) Correct - the upward motion interlock is to prevent raising spent fuel out of the SFP, maintaining the required depth of water over the SF for shielding concerns. (A) Incorrect - the new fuel elevator does travel upward (B) Incorrect -there is no rad interlock with the new fuel elevator (D) Incorrect -this is a 625#

downward

motion stop to prevent damage/binding of a lowering assembly

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Fuel Handling LP	I1-FH-XL-01	II	17	1	6

Material Required for Examination

Question Source: Other Facility

Question Modification Method: Significantly Modified

Question Source Comments: 2001 Prairie Island NRC

Comment Type	Comment
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Record Number: 59 RO Number: SRO Number: 42

Question Fuel Handling Equipment System (FHES)

Unit 1 is in Mode 6 and has commenced core off-load. The following conditions exist:

- 1B EDG is OOS for overhaul
- 1A FHB Exhaust Filter Plenum is aligned and in service
- Containment mini-purge system is in service
- Fuel Handling Building Radiation Monitor, 0RE-AR055, is OOS
- Fuel Handling Building Radiation Monitor, 0RE-AR056, alarm circuitry has just failed. IMD is troubleshooting.

Which of the following describes the required ACTION, if any, to be taken in order to allow core off-load to continue?

- a. No ACTION is required, fuel movements may continue uninterrupted
- b. Core off-load can NOT be conducted until at least one of the FHB rad monitors is repaired
- c. Fuel movement may continue for up to 7 days while restoring one (1) FHB rad monitor to operable status provided 1B FHB Exhaust Filter Plenum is aligned in the Emergency Operating Mode
- d. Fuel movement may be conducted indefinitely provided an appropriate portable monitor is provided and the 1A FHB Exhaust Filter Plenum is aligned in the Emergency Operating Mode

Answer d Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 3 SRO Group: 2

034 Fuel Handling Equipment System

K6. Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling Equipment System:
 K6.02 Radiation monitoring systems 2.6 3.3

Explanation of Answer (A) incorrect - per TRM 3.3.0, with 2 channels inop must place 1 FHB Vent in emergency mode and provide portable monitor or stop fuel movements. (B) Incorrect - may move fuel with vent alignment and protable monitor (C) Incorrect - need to also place portable monitor, and 1B train of FHB Vent has no operable emergency power supply as required by TRM (D) Correct - all actions taken into account.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Fuel Handling LP	I1-FH-XL-01	II	26-32	52	7,10
Tech Requirements Manual	TRM 3.3.o		1,2	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 60 RO Number: 48 SRO Number: 43

Question Main and Reheat Steam System (MRSS)

Prior to start up following completion of A2R09, the Main Steam Isolation Valves (MSIVs) were tested to ensure a closure time of < 5 seconds and that each MSIV actuated to it's isolation position on an actual or simulated actuation signal.

The basis for performing these surveillances was to limit or mitigate ALL of the following EXCEPT:

- a. Accidents that could result in offsite exposures comparable to 10CFR100 limits
- b. The potential for uncontrolled RCS cooldown and positive reactivity restart accident
- c. Total mass and energy release into containment on a HELB
- d. A turbine overspeed condition following a generator trip at power

Answer d **Exam Level** S **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
039 Main and Reheat Steam System
2.2 Equipment Control
2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. 2.5 3.7
Explanation of Answer (A,B&C) Incorrect - Basis for MSIV operability includes all three (D) Correct - no credit is taken to limit or prevent turbine overspeed

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Spec Basis - MSIVs	B 3.7.2	Basis	1-7	0	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type **Comment**

Record Number: 61 **RO Number:** **SRO Number:** 44

Question Main Turbine Generator (MT/G) System

The following conditions exist on Unit 1:

- Reactor power is 80%, steady state
- All systems are in automatic control
- One Main Steam Dump valve, 1MS004A, fails 100% open due to a valve positioner failure.

What is the expected response of the plant due to the steam dump valve failure AND what action can the operator take from the control room to stop the excess steam flow?

- a. Turbine load will decrease by approx. 3% AND reactor power will remain constant. The operator can stop dumping excess steam by taking either Bypass Interlock Switch to OFF/RESET.
- b. Turbine load will remain relatively constant AND reactor power will increase by approx. 3%. The operator can stop dumping excess steam by taking the Steam Dump Mode Selector Switch to STEAM PRESSURE.
- c. Turbine load will decrease by approx. 3% AND reactor power will remain constant. The operator can stop dumping excess steam by taking the Steam Dump Mode Selector Switch to STEAM PRESSURE.
- d. Turbine load will remain relatively constant AND reactor power will increase by approx. 3%. The operator can stop dumping excess steam by taking either Bypass Interlock Switch to OFF/RESET.

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

045 Main Turbine Generator System

A2. Ability to (a) predict the impacts of the following on the Main Turbine Generator System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.08 Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary) 2.8 3.1*

Explanation of then (A) Incorrect - turbine load will remain relatively constant with IMP IN (normal at 100%). Reactor power will

Answer increase due to increased steam flow. (B) Incorrect - selecting Steam Pressure Mode will not close the steam dumps if the failure is in the valve positioner. (C) Incorrect - (see A&B) (D) Correct - turbine load is expected to remain relatively constant with IMP IN, reactor power will increase due to increased steam flow, and either Steam Dump Bypass Interlock switch in OFF/RESET will close all steam dumps (train A&B)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Big Notes	MS-4	Main Steam	1	6	

Material Required for Examination

Question Source: Other Facility

Question Modification Method: Editorially Modified

Question Source Comments: 2001 Prairie Island

Comment Type **Comment**

Record Number: 62 **RO Number:** 49 **SRO Number:** 45

Question Main Turbine Generator (MT/G) System

Once every 31 days, each of the 12 extraction steam nonreturn check valves are tested by observing freedom of movement of the weight arms on each valve. This testing is performed to ensure:

- a. Steam line breaks which occur outside the Auxiliary Building are positively isolated
- b. Flooding does not occur in feedwater heaters, limiting the ability to restart following a reactor trip
- c. Excessive overspeed of the turbine does not occur following a turbine generator trip
- d. Overpressurization of the main condenser does not occur if feedwater heater levels increase too high

Answer C **Exam Level** S **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

045 Main Turbine Generator System

2.2 Equipment Control

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. 2.5 3.7

Explanation of Answer Nonreturn check valves (12) are part of the turbine overspeed protection circuitry and thus protect the turbine from overspeed following a normal turbine trip, specifically from steam flashing in feedwater heaters from reentering the MT. (C) Is correct. (A,B,D) are not discussed and are not part of the basis for turbine overspeed protection.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Specs - TRM	TRM 3.3.g	TSR 3.3.g.2	g-2	1	
Old Tech Spec Basis Doc	B 3/4 3-6	Turb Overspeed	3-6		

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 63 **RO Number:** **SRO Number:** 46 **Number(s)** n

Question Main Feedwater (MFW) System

The following conditions exist on Unit 1:

- Reactor is at 100% power, steady state
- All control systems are in automatic
- Instrument Air is lost to one feedwater regulating valve, 1FRV-510

If no action is taken in response to the FRV, which of the following describes the response of the plant AND followup action required by the MCB operator?

- a. "TURBINE TRIP ABOVE P-8" trips the reactor. All Main Feedwater Pumps AUTOMATICALLY trip. Operator must VERIFY Feedwater Isolation AUTOMATICALLY occurs.
- b. "TURBINE TRIP ABOVE P-8" trips the reactor. All Main Feedwater Pumps must be MANUALLY tripped in 1BwEP ES-0.1. Operator must MANUALLY close all Feedwater Isolation Valves.
- c. "S/G 1A LEVEL LO-2" trips the reactor. All Main Feedwater Pumps AUTOMATICALLY trip. Operator must MANUALLY close all Feedwater Isolation Valves.
- d. "S/G 1A LEVEL LO-2" trips the reactor. All Main Feedwater Pumps must be MANUALLY tripped in 1BwEP ES-0.1. Operator must VERIFY Feedwater Isolation AUTOMATICALLY occurs.

Answer d **Exam Level** R **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

059 Main Feedwater System

A2. Ability to (a) predict the impacts of the following on the Main Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.12 Failure of feedwater regulating valves 3.1* 3.4*

Explanation of Answer (D) Correct - FRVs fail closed on loss of air. SG level will decrease to the lo-2 rx trip setpoint. Lo-2 level does not trip the MFPs. P-4 initiates FW isolation. (A&B) Incorrect - FRV fails closed, levels decrease. (C) Incorrect - Lo-2 level does not trip the MFPs. P-4 does initiate FW Isolation.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Annunciator response proc	BwAR 1-11-A8		1	8E1	
Bwd Big Notes	FW-1		1	4	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 64 **RO Number:** 50 **SRO Number:**

Question Main Feedwater (MFW) System

At 50% power on both units, the Steam Generator programmed level for each unit is:

Unit 1

Unit 2

- a. 33.0%
- b. 50.0%
- c. 60.0%
- d. 81.0%

- 36.3%
- 50.0%
- 63.7%
- 80.8%

Answer c Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems RO Group: 1 SRO Group: 1

059 Main Feedwater System

A3. Ability to monitor automatic operations of the Main Feedwater System including:

A3.02 Programmed levels of the S/G 2.9 3.1

Explanation of Answer Program levels are U-1 at 60.0%, U-2 at 63.3%, from the full range 0 to 100% power. (C) is only correct response

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
SGWLC Lesson Plan	I1-FW-XL-01	II	11	2	2

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 65 RO Number: 51 SRO Number:

The following conditions exist on Unit 1:

- Preparations are underway to perform a reactor startup per 1BwGP 100-2, "Reactor Startup"
- Steam Generator levels are being maintained utilizing tempering line flow via 1FW034A-D and 1FW035A-D
- When testing the reactor trip breakers per step 13 of 1BwGP 100-2, 1FW035D did not automatically close
- 1FW035D was manually closed by the NSO and the condition reported to the Unit Supervisor
- All other feedwater valves responded as designed

Given the above failure, per Tech Specs the reactor startup will ...

- a. Be ALLOWED to continue with BOTH 1FW035D and 1FW034D controlling tempering line flow. Since at least one valve in the line isolated to it's required position, the safety function will not be challenged.
- b. Be ALLOWED to continue, however, 1FW035D must be declared inoperable and closed with power removed from its valve actuator. The Unit is then allowed to operate indefinitely in this configuration.
- c. NOT be allowed to continue because BOTH valves in this line must be OPERABLE to ensure positive isolation and prevent containment out leakage in the event of an accident.
- d. NOT be allowed to continue. In addition to 1FW035D being inoperable, the Aux Relay function of feedwater isolation must be declared inoperable which precludes any future Mode changes until repaired.

Answer b Exam Level S Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1
 059 Main Feedwater System
 2.1 Conduct Of Operations
 2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. 3.4 4.0

Explanation of Answer (A) Incorrect - in order to perserve the safety function, 1FW034D must be closed with power removed - single failure. (B) Correct - 3.6.3 does not require mode reduction if the required action is completed within the time allowed. (C) Incorrect - one valve may be inoperable if closed with power removed. Required action then does not restrict power operations or mode changes (D) Incorrect - the aux relay function has not been affected.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Specs	3.3.2 ESFAS	Funct 5	2-12	A115	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Topic

Record Number: 66 RO Number: SRO Number: 47

Question Auxiliary / Emergency Feedwater (AFW) System

The FIRST signal to automatically start both Aux Feedwater Pumps on each respective unit is received as

steam generator level passes from normal operating level through ____ (1) ____% on Unit 1 and ____ (2) ____% on Unit 2.

____ (1) ____ ____ (2) ____

- a. 12.0 22.8
- b. 18.0 36.3
- c. 23.0 41.3
- d. 88.0 80.8

Answer b Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
Tier: Plant Systems RO Group: 1 SRO Group: 1

061 Auxiliary / Emergency Feedwater System

K1. Knowledge of the physical connections and/or cause-effect relationships between Auxiliary / Emergency Feedwater System and the following:

K1.01 S/G system 4.1 4.1

Explanation of LO-2 SG Levels (2/4) will automatically start both AFW pump on each respective unit. Lo-2 U-1=18.0%,
Answer U-2=36.3%

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Annunciator Response Procs	BwAR 1/2-15-D5		1	8	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 67 RO Number: 52 SRO Number: Number(s) n

The following conditions exist on Unit 1:

- A reactor trip / turbine trip has occurred
- The crew has transitioned out of 1BwEP-0 to 1BwEP ES-0.1, "Reactor Trip Response"
- Step 2 is being performed, "Maintain RCS Temperature Control"
- All steam generator pressures are at 1050 psig and decreasing slowly
- All steam generator narrow range levels are <10%
- RCS temperature is 553°F and decreasing slowly

Which of the following actions is required to control and minimize the cooldown of the RCS?

- a. Maintain maximum AFW flow until steam generator NR levels are >25%, then decrease total AFW flow to 500 gpm
- b. Decrease total AFW flow, maintaining >500 gpm until SG NR levels are >10%, then throttle as needed to control cooldown
- c. Immediately decrease total AFW flow to approximately 25 gpm per SG
- d. Stop the AFW pumps, if operating, and isolate them from the steam generators

Answer b Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1

061 Auxiliary / Emergency Feedwater System

K1. Knowledge of the physical connections and/or cause-effect relationships between Auxiliary / Emergency Feedwater System and the following:

K1.04 RCS 3.9 4.1

Explanation of Answer per 1BwEP ES-0.1, step 2 RNO. Maintain total AFW flow >500 gpm until NR levels are >10% in at least one SG. Then no further restrictions are place on AFW flow rates. (B) Correct (A) Incorrect - 25% level & 500 gpm are too high. (C) Incorrect - cannot reduce AFW flow <500 gpm until NR levels are >10% (D) Incorrect - cannot reduce AFW flow <500 gpm until NR levels are >10%

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor Trip Response & Basis	1BwEP ES-0.1	step 2 RNO	3	100	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type Comment

Record Number: 68 RO Number: 53 SRO Number:

Question A.C. Electrical Distribution System

A loss of all AC power has occurred on Unit 1. The crew is performing 1BwCA-0.0, "Loss of All AC Power", and is preparing to cross-tie to Unit 2 using a limited crosstie to ESF Bus 241. DG 2A is supplying the bus. You have been assigned to monitor ESF Bus amperage as loads are restored on Bus 141.

If ALL of the following loads are started, which ONE will draw the largest running amperage?

- a. 1A MCR Chiller
- b. 1A RCFC
- c. 1A SX Pump
- d. 1A CV Pump

Answer C Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 2

062 A.C. Electrical Distribution

A3. Ability to monitor automatic operations of the A.C. Electrical Distribution including:

A3.01 Vital ac bus amperage 3.0 3.1

Explanation of (A) Incorrect - chiller draws 47 amps (B) Incorrect - RCFC draws 14 amps (Lo speed start as allowed in 0.0)

Answer (C) Correct - 1A SX draws 156 amps (D) Incorrect - 1A CV draws 63 amps

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of All AC - Contingency Procedure	1BwCA-0.0	steps 21,40,18,24		100wog 1C	
Loss of 4KV ESF Bus	1BwOA ELEC-3	step 5 attach A	8	56	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 69 RO Number: 54 SRO Number:

Question A.C. Electrical Distribution System

A reactor trip has just occurred on Unit 1
The automatic bus transfer (ABT) failed to operate for Bus 156

Which of the following loads is now unavailable?

- a. 1A Motor Driven Main Feed Pump
- b. 1A Startup Feedwater Pump
- c. 1A Condensate Pump
- d. 1A Heater Drain Pump

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems RO Group: 2 SRO Group: 2

062 A.C. Electrical Distribution

K2. Knowledge of bus power supplies to the following:

K2.01 Major system loads 3.3 3.4

Explanation of (A) Correct - powered from 156 (B) Incorrect - startup FWP from 159 (C) Incorrect - 1A CD/CB from 159 (D)

Answer Incorrect - 1A HDP from 157

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
AC Distribution LP	I1-AP-XL-01	II	29,30	1	12

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments: 2001 Bwd NRC

Comment Type Comment

Record Number: 70 RO Number: 55 SRO Number: 48

Question D.C. Electrical Distribution System

Which of the following describes how a Reactor Trip Breaker will respond to a LOSS of 125 VDC control power? (Assume the breaker is closed when the loss of control power occurs)

- a. Trips OPEN due to loss of power to the SHUNT coil.
- b. Trips OPEN due to loss of power to the UNDERVOLTAGE coil
- c. is NOT capable of tripping on a SHUNT trip
- d. is NOT capable of tripping on a UNDERVOLTAGE trip

Answer c Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 1

063 D.C. Electrical Distribution

K2. Knowledge of bus power supplies to the following:

K2.01 Major dc loads 2.9* 3.1*

Explanation of Answer A. Incorrect because the shunt coil is normally de-energized. B. & D. incorrect because the undervoltage coil is supplied with 48v power from SSPS

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Electrical Prints	20E-1-4030-RD6	N/A	1	P	
Solid State Protection System	I1-RP-XL-04	II	9,17	0	4,10

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method:

Question Source Comments: 2000 Bwd NRC. 1998 Calloway NRC Exam

Comment Type	Comment
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Record Number: 71 RO Number: 56 SRO Number: 49

Question Emergency Diesel Generator (ED/G) System

When synchronizing an Emergency Diesel Generator to an energized ESF bus, immediately after closing the generator output breaker, load the EDG to 500KW by going to _____(1)_____ on the Diesel Generator _____(2)_____.

- | | |
|-----------------|-------------------------|
| _____ (1) _____ | _____ (2) _____ |
| a. Raise | Governor Adjust Control |
| b. Raise | Voltage Adjust Control |
| c. Lower | Governor Adjust Control |
| d. Lower | Voltage Adjust Control |

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
064 Emergency Diesel Generators

A1. Ability to predict and/or monitor changes in parameters associated with operating the Emergency Diesel Generators controls including:

A1.08 Maintaining minimum load on ED/G (to prevent reverse power) 3.1 3.4

Explanation of Answer (B) is correct. (A) Incorrect - lowering DG Speed will decrease load, approaching the reverse power trip setpoint (C&D) are incorrect - adjusting the voltage control will not affect DG loading

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Diesel Generator Startup	BwOP DG-11	F	13	23	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 72 **RO Number:** 57 **SRO Number:** 50

Question Liquid Radwaste System (LRS)

What TWO conditions will INDEPENDENTLY cause automatic closure of Liquid Radwaste Release Tank

Discharge Key Locked Valve 0WX353?

- a. Low circulating water blowdown flow and high radiation sensed in the CW blowdown flow
- b. Low circulating water blowdown flow and high radiation sensed in the release header
- c. High release header flow and high radiation sensed in the release header
- d. High release header flow and high radiation sensed in the CW blowdown flow

Answer b Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems RO Group: 1 SRO Group: 1

068 Liquid Radwaste System

A4. Ability to manually operate and/or monitor in the control room:

A4.04 Automatic isolation 3.8 3.7

Explanation of Answer Per BwOP WX-526T1, (B) is only correct combination provided

Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Liquid radwaste release form	BwOP WX-526T1	E,G	22,33	18	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: 1999 Bwd NRC

Comment Type Comment

Record Number: 73 RO Number: 58 SRO Number: 51

Question Liquid Radwaste System (LRS)

An operator spent 30 minutes in a field of 150 mr/hour lining up to transfer the contents of one liquid radwaste monitor tank to another. He said later that if he had 'preplanned' his work he could have been finished in 20 minutes. How much dose could have been avoided if he had preplanned the job?

- a. 50 mrem
- b. 25 mrem
- c. 12.5 mrem
- d. 10.5 mrem

Answer b Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems RO Group: 1 SRO Group: 1

068 Liquid Radwaste System

K5. Knowledge of the operational implications of the following concepts as they apply to the Liquid Radwaste System:

K5.03 Units of radiation, dose, and dose rate 2.6 2.6

Explanation of 150mrem / 60min x 20min = 50mrem if done in 20 minutes. Savings of 75-50=25 mrem (B) Correct

Answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Health Physics / NGET Rad Protection LP		Practice problems	1-35		8

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment
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Record Number: 74 RO Number: 59 SRO Number:

Question Waste Gas Disposal System (WGDS)

Which of the following REDUCES the possibility of an unintentional radioactive release to the atmosphere from a Waste Gas Decay Tank (WGDT) relief valve lifting?

- a. OGW014, Waste Gas Discharge valve, will close automatically on detected high radiation in the discharge header, isolating the relief path
- b. WGDT relief valves discharge directly to the vent header so that flow is directed from the online tank directly to the standby tank
- c. The Waste Gas Compressor discharge pressure is automatically limited to less than the WGDT relief valve pressure setpoint
- d. WGDT inlet valve closes automatically on high pressure isolating the on-line WGDT and directing flow

to the standby WGDT

Answer d Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 1 SRO Group: 1

071 Waste Gas Disposal System

K3. Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following:

K3.05 ARM and PRM systems 3.2 3.2

Explanation of (D) Correct - tanks are automatically switched on high pressure. (A) incorrect - all reliefs discharge downstream of 0WX014 so the relief path is not isolated (B) Incorrect - discharge is to the plant vent (C) incorrect - discharge of the compressor has no affect on the suction (tank header)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Gaseous Radwaste LP	11-GW-XL-01	II	12-14	0	6,10
Bwd Big Notes	RW-1			0	

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Editorially Modified

Question Source Comments: 1999 Bwd NRC

Comment Type Comment

Record Number: 75 RO Number: 60 SRO Number: 52

Question Area Radiation Monitoring (ARM) System

Radiation levels in the Fuel Handling Building INCREASED causing BOTH Fuel Handling Incident radiation monitors (AR055 and AR056) to simultaneously reach their actuation setpoints.

Which of the following would AUTOMATICALLY occur due to this condition?

- a. B Train FHB Charcoal Booster Fan starts, then A Train FHB Charcoal Booster Fan starts.
- b. B Train FHB Charcoal Booster Fan will start ONLY if A Train has failed to start.
- c. A Train FHB Charcoal Booster Fan starts, then B Train Charcoal Booster Fan starts.
- d. A Train FHB Charcoal Booster Fan will start ONLY if B Train has failed to start.

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

072 Area Radiation Monitoring System

A3. Ability to monitor automatic operations of the ARM system including:

A3.01 Changes in ventilation alignment 2.9* 3.1

Explanation of Answer A. Incorrect. Damper interlocks prevent both trains from starting. B Train gets a start signal first. When it starts, it's dampers position, an interlock preventing the start of A Train. B. Incorrect - it is the reverse of D, the correct answer. C. Incorrect - B gets the start signal first. D. Correct

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Norse Notes Aux Bldg Vent	VA-2	FHB Interlocks			
System LP CH 43A			11,34,35		

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: 2001 Bwd NRC

Comment Type **Comment**

Record Number: 76 **RO Number:** 61 **SRO Number:** 53

Question Area Radiation Monitoring (ARM) System

The detector for 1RT-AR011J, Containment Fuel Handling Incident Train A Rad Monitor, has failed causing the output of the monitor to go high.

Which of the following automatic actions will occur as a result of this failure?

- a. OVA04CA, Fuel Handling Charcoal Booster Fan is started
- b. 1VQ004A, Containment Mini-Flow Purge Supply Isolation Valve is closed
- c. 1VQ003, Post LOCA Charcoal Filter Isolation Valve is opened
- d. 1VQ003C, Post LOCA purge exhaust fan is started

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

072 Area Radiation Monitoring System

K4. Knowledge of ARM system design feature(s) and or interlock(s) which provide for the following:

K4.01 Containment ventilation isolation 3.3* 3.6*

Explanation of Answer (A) Incorrect - this fan is started via AR055&56 skids, not AR11J or 12J (B) Correct - this is part of the cnmt isolation signal generated. (C&D) Incorrect - these receive no auto actuation signal from any rad monitor.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RM-11 annunciator response	BwAR 4-1AR011J	B	1	2	
Bwd Big Notes - Cnmt Purge	VP-2			5	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment

Record Number: 77 **RO Number:** 62 **SRO Number:** 54

The following conditions exist on Unit 1:

- A loss of all Circulating Water Pumps has occurred due to excessive grass collection in the intake bay.
- A reactor trip / turbine trip was manually initiated by the operators.
- During performance of 1BwEP-0, a SGTR occurred on the 1B steam generator
- The crew transitioned to and performed actions contained in 1BwEP-3, "Steam Generator Tube Rupture"
- The RCS cooldown and depressurization steps to equalize RCS and ruptured SG pressure have been completed
- SI was terminated and the crew is now investigating the appropriate post-SGTR cooldown method to use
- While investigating cooldown options, RCS subcooling was lost
- Additional ECCS pumps have been started and aligned, but subcooling is not recovering

Which of the following procedures must be used to continue the post SGTR cooldown and recovery actions from this point?

- a. 1BwEP-3 ,"SGTR" must be continued until conditions exist for establishing RHR shutdown cooling
- b. 1BwEP ES-3.1, "Post-SGTR Cooldown Using Backfill" must be used quickly to recover Pzr level and subcooling
- c. 1BwEP ES-3.3, "Post-SGTR Cooldown Using Steam Dumps" must be used as this is the preferred method of recovery
- d. 1BwCA-3.1, "SGTR With Loss of Reactor Coolant, Subcooled Recovery Desired" is the only option available and must be implemented

Answer d Exam Level S Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Plant Systems RO Group: 2 SRO Group: 2
 075 Circulating Water System
 2.4 Emergency Procedures / Plan
 2.4.6 Knowledge symptom based EOP mitigation strategies. 3.1 4.0

Explanation of Answer (A) Incorrect - step 38, Go to Appropriate Post-SGTR Cooldown Method, is the LAST step in EP-3. There is no continuation from here without a transition. (B) Incorrect - OAS and step 2 of ES-3.1 require transition to CA-3.1 with the loss of subcooling. (C) Incorrect - same as B. and with loss of CW, no condenser vacuum exists to use steam dump system. (D) Correct - as required by each procedure's OAS and step 2 in each post-SGTR cooldown procedure.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
EOPS - SGTR	1BwEP-3	step 38 & OAS	45	100	
Post-SGTR Cooldown procedures	1BwEP ES-3.1 & 3.3	step 2 & OAS	3	1A	
EP-3 Basis	Background Docs	ES-3.1 step 2	21	1C	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Topic Comment

Record Number: 78 RO Number: SRO Number: 55

Question Fire Protection System (FPS)

Which of the following Fire Protection subsystems is used to provide coverage for the 1B Auxiliary Feedwater Pump?

- a. Foam
- b. Water
- c. Halon
- d. CO2

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Plant Systems RO Group: 2 SRO Group: 2

086 Fire Protection System

K1. Knowledge of the physical connections and/or cause-effect relationships between Fire Protection System and the following:

K1.03 AFW System 3.4* 3.5*

Explanation of Answer (D) Correct. It is the only FP system in the 1B AFW Room

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Big Notes	FP-2			1	
Fire Protection LP	I1-FP-XL-01	II			

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 79 RO Number: 63 SRO Number: 56

Number(s) n

Question Fire Protection System (FPS)

In which ONE of the following areas is water used as the PRIMARY fire suppression agent?

- a. MPT/UAT/SAT Transformers
- b. Upper Cable Spreading Room
- c. Lower Cable Spreading Room
- d. Diesel Generator Rooms

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

086 Fire Protection System

K5. Knowledge of the operational implications of the following concepts as they apply to the Fire Protection System:

K5.03 Effect of water spray on electrical components 3.1 3.4

Explanation of water is not used where damage may result from spray on equipment. (A) is correct answer (B) primary agent

Answer is halon (C) primary agent is CO2 (D) primary agent is CO2

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Big Notes - Fire Protection	FP-2			1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 80 **RO Number:** 64 **SRO Number:** 57

Question Dropped Control Rod

The following plant conditions exist on Unit 1:

- Reactor power is 75%
- Tave is 565 °F
- Pressurizer pressure is 2235 psig
- Rod H-8 drops to the bottom of the core
- ROD AT BOTTOM annunciator is LIT

The INITIAL primary plant response to this event is RCS pressure _____(1)_____ and RCS Tave _____(2)_____.

_____ (1) _____ (2) _____

- a. Increases Increases
- b. Decreases Increases
- c. Increases Decreases
- d. Decreases Decreases

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 1

Dropped Control Rod

AA1. Ability to operate and / or monitor the following as they apply to Dropped Control Rod:

AA1.06 RCS pressure and temperature 4.0 4.1

Explanation of Answer RCS temperature and pressure will decrease with power immediately following a dropped control rod. (D) is Correct.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Dropped or Misaligned Rod	1BwOA ROD-3	Symptoms	1	101	
BwOA ROD-3 Lesson Plan	I1-OA-XL-34	ii	4	8	2,5

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 82 RO Number: 66 SRO Number: 59

Question Inoperable/Stuck Control Rod

The following plant conditions exist on Unit 1

- PDMS is inoperable
- Control Bank D, Rod D-12 has become misaligned from the rest of the group by 10 steps
- Thermal power is 100% and stable
- QPTR associated with N41 has just been determined to be 1.10

If QPTR cannot be reduced to less than 1.10 over the next 48 hours, thermal power will be limited to:

- a. 0%
- b. 50%
- c. 70%
- d. 77%

Answer b Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1
 005 Inoperable/Stuck Control Rod

AK1. Knowledge of the operational implications of the following concepts as they apply to Inoperable/Stuck Control Rod:
 AK1.01 Axial power imbalance 3.1 3.8

Explanation of Answer Per 3.2.4 - reduce power greater than or equal to 3% for each 1% over 1.00 QPTR will be measured Once per 12 hours and thermal power reduced within 2 hours of each determination. (A) is incorrect - the TS is not applicable below 50% power. (C&D) are incorrect, power levels listed are too high. 77% is only a 3% reduction.

70% is only the initial reduction.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Dropped or Misaligned Rod	1BwOA Rod-3		8,9		101
Tech Specs	3.2.4		3.2.4-1		110

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment

Record Number: 83 RO Number: 67 SRO Number: 60

Question Reactor Trip

An automatic reactor trip has occurred requiring entry into 1BwEP-0, "Reactor Trip or Safety Injection". During performance of the first step, the operator cannot readily ascertain if the Reactor Trip and Bypass Breakers are open. All Rod Bottom lights are LIT and all Nuclear Instrumentation indicates neutron flux is rapidly decreasing with a -0.3 DPM startup rate.

What is the NEXT action required of the operators?

- a. Manually trip the reactor
- b. Verify the turbine is tripped
- c. Send an operator to locally verify reactor trip breakers are open
- d. Transition to 1BwFR-S.1, "Response to Nuclear Power Generation/ATWS"

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
007 Reactor Trip

EK2. Knowledge of the interrelations between Reactor Trip and the following:

EK2.03 Reactor trip status panel 3.5 3.6

Explanation of Answer Per 1BwEP-0 Step 1, actions are closed bulleted therefore reactor trip breakers must be verified open or the RNO applied to manually trip the reactor. After the manual trip attempt the operators may proceed to step 2 (B incorrect). Transition to FR-S.1 is not required with SUR more negative than -.02 DPM. (D incorrect). No provisions are allowed to dispatch an operator to locally verify the status of Reactor Trip Breakers while performing the RNO of step 1 (C incorrect).

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactor Trip or Safety Injection	1BwEP-0	Step 1	3	100WO G1C	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: 1997 Bwd NRC Exam

Comment Type **Comment**

Record Number: 84 **RO Number:** 68 **SRO Number:** 61

Question Small Break LOCA

Following a small break LOCA, some reactor decay heat might be removed by "reflux flow".

Reflux flow is best described as:

- a. Steam produced inside the core is condensed in the steam generator tubes and returned to the core via gravity counterflow along the bottom of each partially filled hot leg pipe.
- b. Steam produced inside the core is condensed in the steam generator tubes and returned to the core via natural circulation flow along the bottom of each cold leg pipe.
- c. Liquid heated by the core is subsequently cooled inside the steam generator tubes and returned to the core via counterflow along the top of each partially filled hot leg pipe.
- d. Liquid heated by the core is subsequently cooled inside the steam generator tubes and returned to the core via natural circulation flow along the bottom of each cold leg pipe.

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 2
 009 Small Break LOCA

EK1. Knowledge of the operational implications of the following concepts as they apply to Small Break LOCA:

EK1.01 Natural circulation and cooling, including reflux boiling 4.2 4.7

Explanation of (A) Correct - reflux cooling involves the condensation of steam in the steam generators and draining the
 Answer resultant liquid back into the Rx core via the hot leg. Occurs after core voiding disrupts natural circulation flow

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Background Documents	E-1	LOCAs	16	1	
Mitigating Core Damage	MTG	LOCA Core Cooling	1-1-89/90		

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment

Record Number: 87 RO Number: 69 SRO Number: 64

Question Large Break LOCA

Restoration of adequate cooling flow to the core during a large break LOCA is best achieved by:

- a. Starting all Reactor Coolant Pumps
- b. Establishing high-head Safety Injection flow
- c. Reducing RCS pressure by opening both Pzr PORVs
- d. Rapidly depressurizing all Steam Generators to atmospheric pressure

Answer b Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 1

011 Large Break LOCA

EA2. Ability to determine and interpret the following as they apply to Large Break LOCA:

EA2.10 Verification of adequate core cooling 4.5 4.7

Explanation of "Reinitiation of High Head SI is the most effective method to recover the core and restore adequate core

Answer cooling" 1BwFR C.1 background document. (B) is the correct response.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Background Documents - Inadequate cooling	BwFR C.1	2	2	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 88 RO Number: 70 SRO Number:

Question Large Break LOCA

Which of the following explains why it is necessary to start Auxiliary Feedwater and verify flow during a large break LOCA accident?

- a. To provide a positive static head of water to prevent steam generator tube leakage.
- b. To remove RCS decay heat via forced circulation coolant flow.
- c. To remove RCS decay heat via natural circulation coolant flow.
- d. To provide a secondary heat sink for Post LOCA Cooldown and Depressurization

Answer a Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 1

011 Large Break LOCA

EK3. Knowledge of the reasons for the following responses as they apply to Large Break LOCA:

EK3.03 Starting auxiliary feed pumps and flow, ED/G, and service water pumps 4.1 4.3

Explanation of Answer Per background docs - (A) Correct. Since SG's will eventually be depressurized, water level will prevent primary to secondary leakage. (B,C,D) Incorrect - steam generators are not required as a heat sink for large break LOCAs (temps and pressures typically remain higher than in the RCS)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Reactor or Secondary Coolant	1BwEP-1	Step 3	4	100	
Background Documents	EP-1	Step 3	51	1C	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 89 RO Number: 71 SRO Number: 65

Question Reactor Coolant Pump (RCP) Malfunctions (Loss of RC Flow)

Given the following:

- Unit 1 is operating at 100% power
- RCP No. 1 SEAL LEAKOFF FLOW HIGH alarm is received
- No. 2 seal leakoff high flow alarm has been printed
- RCP No. 1 seal leakoff recorder indication is offscale high on the HIGH range

Which of the following has occurred and what action is procedurally directed to be taken?

- a. The No. 1 and No. 2 RCP seals have failed and a controlled reactor shutdown is required
- b. ONLY the No. 2 RCP seal has failed and continued monitoring of RCP conditions is required
- c. ONLY the No. 1 RCP seal has failed and an immediate reactor trip is required
- d. The No. 2 and No. 3 RCP seals have failed and continued monitoring of RCP conditions is required

Answer c **Exam Level** B **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

017 Reactor Coolant Pump Malfunctions (Loss of RC Flow)

AA1. Ability to operate and / or monitor the following as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow):

AA1.22 RCP seal failure/malfunction 4.0 4.2

Explanation of Answer Indications are that the No. 1 seal has failed. The operator action summary of 1BwOA RCP-1 states to go to step 12 which states to trip the reactor and the RCP. Due to the high seal leakoff flow, continued monitoring is not the proper action to take. A controlled RCP is required if seal leakoff is high but not in alarm. #3 seal has not been affected.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
RCP Seal Failure	1BwOA RCP-1	OAS			
RCS LP	AP-XL-01				8
RCP Seal Failure LP	I1-OA-XL-27	II	4	10	3

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments: 1999 Bwd NRC

Comment Type **Comment**

Record Number: 90 **RO Number:** 72 **SRO Number:** 66

Question Emergency Boration

Per the TRM, which of the following conditions meets the associated MINIMUM requirement for the Boric Acid Storage System to be considered OPERABLE in Mode 3?

- a. A contained borated water level of 35%
- b. A boron concentration of 6800 ppm
- c. A solution temperature of 69°F
- d. A flowpath to the CV pump via 1CV110A, Boric Acid to Blender Vlv

Answer c Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

024 Emergency Boration

AK1. Knowledge of the operational implications of the following concepts as they apply to Emergency Boration:

AK1.04 Low temperature limits for boron concentration 2.8 3.6

Explanation of Answer (C) Correct - TRM requires 40% level, 7000 ppm, and 65°F. (A&B) Incorrect. (D) Incorrect - surveillance for flowpath includes 112D&E and 1CV8104

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Reactivity Control Systems	TRM	3.1.f	2	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 93 RO Number: 74 SRO Number: 68

Question Loss of Residual Heat Removal System (RHRS)

Given the following plant conditions on Unit 1

- Unit 1 is shutdown with B train of RHR providing shutdown cooling
- RCS Pressure is 350 psig
- RCS Tave is 330°F
- RCS Cooldown rate is 30°F/hr
- RHR total flow is 3300 gpm
- 1RH607, 1B RH Heat Exchanger Flow Control Valve, is throttled 52% open (1500 gpm)

Flow transmitter 1FT-619, RHR Discharge Flow, fails LOW with the flow controller for 1RH619 in AUTOMATIC.

What further indications will occur as a result of this failure?

- a. The RCS Cooldown rate and CCW temperatures will both INCREASE
- b. The RCS Cooldown rate will INCREASE and CCW temperatures will DECREASE
- c. The RCS Cooldown rate and CCW temperatures will both REMAIN THE SAME
- d. The RCS Cooldown rate and CCW temperatures will both DECREASE

Answer d Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 2

025 Loss of Residual Heat Removal System

AK2. Knowledge of the interrelations between Loss of Residual Heat Removal System and the following:

AK2.03 Service water or closed cooling water pumps 2.7 2.7

Explanation of (D) Correct - with 1FT619 failing low, more flow will be demanded from flow control valve 619, more flow will bypass the RH Heat Exchanger, less RCS flow through the heat exchanger will decrease the RCS cooldown rate. Less heat is transferred to CCW and CCW temps decrease.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Normal Operating Procedures - RH Cooling	BwOP RH-6	F	14-15	26	
Bwd Big Notes	RH-1 RHR Cooldown		1	3	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 94 RO Number: 75 SRO Number: 69

Question Loss of Component Cooling Water (CCW)

The following conditions exist on Unit 1:

- A normal plant shutdown is in progress per 1BwGP 100-5, Plant Shutdown and Cooldown
- Train A of RH cooling was placed in service 5 minutes ago
- 3 minutes ago the following alarms were received:
 - Annunciator 1-7-E3, "RCP THERM BARR CC WTR TEMP HIGH"
 - Annunciator 1-7-E5, "RCP BRNG CC WTR TEMP HIGH"
 - Annunciator 1-2-C5, "CC HX OUTLET TEMP HIGH"
- The following readings exist on all running RCPS:
 - Motor bearing temperatures are 165°F
 - Lower radial bearings are 170°F
 - Seal outlet temperatures are 135°F

Operator action in response to these conditions will be to _____(1)_____ because _____(2)_____.

_____ (1) _____

_____ (2) _____

- | | |
|---------------------------------------|--|
| a. Immediately stop all running RCPS | RCP bearing temperature limits have been exceeded due to a loss of cooling flow |
| b. Reduce the RCS cooldown rate | CCW heat exchanger temperatures are approaching design limits allowed for RCS cooldown |
| c. Manually actuate SI, enter 1BwEP-0 | A loss of ALL Component Cooling Water has occurred on Unit 1 |
| d. Start additional CCW pumps | More flow is required through the CC Heat Exchanger to control CCW temperatures |

Answer b Exam Level S Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

026 Loss of Component Cooling Water

AA2. Ability to determine and interpret the following as they apply to Loss of Component Cooling Water:

AA2.04 The normal values and upper limits for the temperatures of the components cooled by SWS 2.5 2.9*

Explanation of Answer (A) Incorrect - RCP bearing temperatures are well within limits. Motor bearings <195°F, Lower radial bearing <225°F, Seal outlet <235°F. (B) Correct - per 1BwOA PRI-6, with CC suction temp and discharge temps in alarm, heat exchanger outlet will be >120°F which is max allowed by TS (Basis section 3.7.7-3). Reduce the cooldown rate in the RCS. (C) Incorrect - No SI criteria has been met, a total loss of CC is not occurring. (D) Incorrect - increasing CC flow through one heat exchanger will only serve to increase the RCS cooldown - contrary to actions required in OA-PRI-6.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Component Cooling Malfunction	1BwOA PRI-6	Main Body	6,7	100	
Annunciator Response Procedures	1-2-C5&D5, 1-2-E3&E5	Cause, Actions	1	vari	
Tech Specs	Basis	3.7.7	7-3	0	

Topic Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type **Comment**

Record Number: 95 **RO Number:** **SRO Number:** 70

Number(s) n

Question Loss of Component Cooling Water (CCW)

Unit 1 is operating at 100% reactor power, steady state conditions. All controlling systems are operating normally in automatic. Operators are performing steps in 1BwOA PRI-6, "Component Cooling Malfunction" due to a slowly lowering level in BOTH halves of the CC surge tank when the following sequence of annunciators is received:

- 1-2-E4, "CC SURGE TANK AUTO-M/U ON"
- 1-2-A5, "CC SURGE TANK LEVEL HIGH LOW"
- 1-2-A4, "CC PUMP TRIP"
- 1-7-A/B/C/D4, "RCP 1A/B/C/D THERM BARR CC WTR FLOW LOW"

The NEXT procedure that must be entered by the operators is _____(1)_____ because _____(2)_____.

_____ (1) _____

_____ (2) _____

- | | |
|--|---|
| a. BwOP CC-5 Component Cooling Water Make-up | Auto make-up to the surge tank has failed and must be restored |
| b. 1BwOA RCP-2 Loss of Seal Cooling | RCP seal failures are imminent due to the loss of thermal barrier cooling |
| c. 1BwEP-0 Reactor Trip or Safety Injection | The reactor must be manually tripped and all RCPs stopped immediately |
| d. 1BwCA-0.0 Loss of All AC Power Unit 1 | All ECCS and safe shutdown loads must be stopped/prevented from starting |

Answer c	Exam Level S	Cognitive Level Application	Facility: Braidwood	ExamDate: 7/17/02
Tier: 026	Emergency and Abnormal Plant Evolutions	RO Group: 1	SRO Group: 1	
2.4	Loss of Component Cooling Water			
2.4.4	Emergency Procedures / Plan			
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.			4.0 4.3

Explanation of Answer (C) Correct - symptoms are of decreasing surge tank level and loss of all running CC pumps. Operators are directed to enter PRI-6 for the CC malfunction, and EP-0 if the surge tank decreases to <13% to trip the reactor and stop all RCPs. (A) Incorrect - leakage may be more than auto makeup can recover. Condition is unknown at this time. Of immediate concern is loss of CC to RCPs, making this a low priority. (B) Incorrect - the loss of thermal barrier cooling is not a concern as long as seal injection is maintained. (D) Incorrect - ECCS / safe shutdown loads are cooled by SX

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Component Cooling Malfunction	1BwOA PRI-6	Attachment A	10	100	
Annunciator Response	BwAR 1-2-A4,A5,E4	Operator Actions	1		

Material Required for Examination

Question Source: New

Question Modification Method:

Question Fuel Handling Incidents

The reason for limiting the maximum load to 2000 lbs. traveling over the fuel assemblies in the Spent Fuel

Pool is:

- a. To NOT exceed the lift capacity of the FHB crane
- b. To ensure spent fuel racks are protected from excessive lifting forces
- c. To limit the magnitude of a potential radioactive release
- d. To limit the potential flooding of the spent fuel pool ventilation system

Answer c **Exam Level** S **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

036 Fuel Handling Incidents

AA2. Ability to determine and interpret the following as they apply to Fuel Handling Incidents:

AA2.03 Magnitude of potential radioactive release 3.1* 4.2

Explanation of Answer Per TRM 3.9.d . (Old TS 3.9.7) Crane travel with loads in excess of 2000 lbs is limited to ensure in the event the load is dropped, the activity release will be limited to that contained in a single fuel assembly and possible distortion of or fuel in the racks will not result in a critical array.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Refueling Operations	TRM	3.9.d	1	1	
TS (old) Basis	TS 3/4 9-2	Basis	9-2	A25	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 98 **RO Number:** **SRO Number:** 73

Question Steam Generator (S/G) Tube Leak

The following conditions existed on Unit 1:

- 100% reactor power
- Small Steam Generator Tube Leak (5 gpd) on 1A Steam Generator
- A shutdown has been ordered to repair the leak

If the Main Turbine were to trip, what is the MAXIMUM power level that the turbine could trip from that would result in the least amount of direct radioactive release to the environment?

- a. 40%
- b. 60%
- c. 80%
- d. 100%

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
037 Steam Generator Tube Leak

AK3. Knowledge of the reasons for the following responses as they apply to Steam Generator Tube Leak:

AK3.09 Maximum load change capability of facility 2.7* 3.1

Explanation of Answer A. Correct. Steam dumps will absorb a 40% load rejection, which is essentially the situation in question. 10% more can be absorbed by the rods (10 + 40 = 50) but that is not a distractor. Anything higher will result in opening of the SG PORV's.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Horse Notes - Steam Dumps	MS-4 Main Steam Dumps Purpose			6	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: 2001 Bwd NRC

Comment Type	Comment

Record Number: 99 **RO Number:** 77 **SRO Number:** 74

Question Steam Generator Tube Rupture (SGTR)

A SGTR was in progress on Unit 1, and the control room operators were performing 1BwEP-3, "Steam Generator Tube Rupture". The operators identified and isolated the ruptured Steam Generator, and they cooled down and depressurized the RCS. When conditions were established that indicated Safety Injection flow was no longer required, the operators were directed to stop all but one CV pump and both SI pumps. 1B CV Pump and 1A SI Pump were successfully stopped.

When stopping the 1B SI Pump, the control switch indicated a GREEN (after trip) target, but positive indications of pump amps and discharge pressure went unnoticed by the operator. What effect will this have on continued operations if the status of the SI pump remains undetected?

- a. The ruptured S/G will eventually fill with water, and the atmospheric relief valve will lift.
- b. The RCS will quickly repressurize and experience an overpressure transient.
- c. Excessive cooldown of the RCS will occur, possibly causing a PTS concern in the RCS.
- d. Damage to the SI pump will occur due to overheating from insufficient flow through the pump

Answer a **Exam Level** B **Cognitive Level** Comprehension **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

038 Steam Generator Tube Rupture

EA1. Ability to operate and / or monitor the following as they apply to Steam Generator Tube Rupture:

EA1.24 Safety injection pump ammeter and indicators 3.6* 3.4

Explanation of Answer (A) Correct - per the reference document, SI must be terminated when conditions are reached in order to prevent SG overfill. (B) Incorrect - with only 1 SI pump the repressurization would be slow, and only reach the shutoff head of the SI pump which is ~1500 psig. (C) Incorrect - PTS is not a credible concern with all RCPs running. The amount of RCS cooldown at the time of SI termination is set by the operators use of steam release and not dependent on SI flow. (D) Incorrect - SI pumps have recirc lines open to the RWST.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
WOG background documents	1BwEP-3				

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type	Comment

Record Number: 100 **RO Number:** 78 **SRO Number:** 75

Question Steam Generator Tube Rupture (SGTR)

Using the Main Steam radiation monitors and Figure 2 of 1BwOS SG-1, "Steam Generator Primary to Secondary Leakage Estimation", what MINIMUM change in dose rate is necessary to cause the Unit to exceed the Tech Spec limit for one (1) steam generator tube leakage requiring a unit shutdown?

- a. 0.05 mr/hr
- b. 0.10 mr/hr
- c. 0.15 mr/hr
- d. 0.20 mr/hr

Answer b **Exam Level** S **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

038 Steam Generator Tube Rupture

EA2. Ability to determine and interpret the following as they apply to Steam Generator Tube Rupture:

EA2.11 Local radiation reading on main steam lines 3.7* 3.9*

Explanation of Answer TS Limit for SG Tube Leakage is 150 gpd through any 1 SG. (B) Correct - a .1 mr/hr increase over background will yield an estimated leak rate of 150 gpd per 1BwOS SG-1, Fig 2. (A) Incorrect - yields about 75 gpd est. (C) Incorrect - >than the minimum of B at 225 gpd (D) Incorrect - >than the minimum of B at 300 gpd

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Steam Generator Leakage Estimation	1BwOS SG-1	F, Figure 2	4	4	
Tech Specs - Operational Leakage	2.4.13	LCO	3.4.13-1	A98	

Material Required for Examination 1BwOS SG-1SG Leakage Estimation - Figure 2 (page 12)

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type **Comment**

Record Number: 101 **RO Number:** **SRO Number:** 76

Question Steam Line Rupture

A reactor trip and safety injection has occurred on Unit 1. The following conditions exist:

- RCS Tave is 485°F and decreasing
- RCS Pressure is 1300 psig and decreasing
- Containment pressure is 0.3 psig and stable
- Containment rad monitors are Green
- Aux Building rad monitors are Green
- Steam Generator Parameters:

SG:	A	B	C	D
Pressure	1000 psig stable	1000 psig stable	450 psig decreasing	1000 psig stable
NR Level	30% increasing	28% increasing	0% (no trend)	30% increasing
MS Rad	Green	Green	Green	Green

In addition to 1BwEP-0, "Reactor Trip or Safety Injection", the above indications are entry level conditions for which other Emergency Operating Procedure?

- a. 1BwCA-1.2 LOCA Outside Containment
- b. 1BwEP-1 Loss of Reactor or Secondary Coolant
- c. 1BwEP-2 Faulted Steam Generator Isolation
- d. 1BwCA-2.1 Uncontrolled Depress of all Steam Generators

Answer c Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1
 040 Steam Line Rupture
 2.4 Emergency Procedures / Plan
 2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. 4.0 4.3

Explanation of Answer (C) Correct - 1C SG pressure is decreasing in an uncontrolled manner. (B) Incorrect - cnmt rad and pressure are normal post trip readings. (A) Incorrect - parameters do not support the entry conditions. AB Rad is normal (D) parameters do not support the entry conditions. SG's A,B&D are stable

Reference Title	Facility Reference Number	Section	Page	Revisio L. O.
Reactor Trip or Safety Injection	1BwEP-0		22,23	100wog 1C

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 102 RO Number: 79 SRO Number:

Question Loss of Main Feedwater (MFW)

The control room operators are responding to a RED condition on the heat sink status tree. While they attempt to restore feed flow to a S/G, conditions degrade to the point that RCS bleed-and-feed must be established.

The reason RCS bleed and feed must be established QUICKLY is to prevent:

- a. Inability to provide sufficient injection flow for core cooling due to high RCS pressure
- b. High temperature and pressure failure of Steam Generator tubes
- c. An overpressurization challenge to the reactor vessel
- d. A rapid RCS overpressurization, followed by a rapid RCS depressurization due to RCP seal failures.

Answer a **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
054 Loss of Main Feedwater

AA1. Ability to operate and / or monitor the following as they apply to Loss of Main Feedwater:

AA1.04 HPI, under total feedwater loss conditions 4.4 4.5

Explanation of Answer (A) Correct - per H.1 background documents. Early bleed and feed allows maximum RCS pressure drop, greater SI flow rates and ensures effective heat removal. The further the transient is allowed to progress before

bleed and feed is initiated, the smaller the initial depressurization will be, lower SI flow rates, greater repressurization and higher net inventory losses.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Functional Restoration Procedures	1BwFR-H.1	OAS, Step 3	3	100	
Background Documents	1BwFR-H.1	Bleed & Feed	34,35	1	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: 1998 Seabrook NRC

Comment Type	Comment
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Record Number: 103 **RO Number:** 80 **SRO Number:** 77

Question Loss of Offsite and Onsite power (Station Blackout)

The following conditions exist:

- A station blackout has occurred
- 1A EDG tripped on differential overcurrent
- 1B EDG failed to field flash
- NO unit SAT's are energized
- Both Unit 2 EDG's were successfully started and are carrying buses 241 and 242
- Unit 2 has determined that BOTH buses 241 and 242 are available for crosstie

Given the available AC sources, what is the preferred method for restoration of AC power on Unit 1:

Cross-tie ESF bus ____ (1) ____ and verify ____ (2) ____ loads on Unit 2 RUNNING.

____ (1) ____ ____ (2) ____

- a. 241 to 141 Train A
- b. 241 to 141 Train B
- c. 242 to 142 Train A
- d. 242 to 142 Train B

Answer b Exam Level S Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

055 Station Blackout

EA2. Ability to determine and interpret the following as they apply to Station Blackout:

EA2.03 Actions necessary to restore power 3.9 4.7

Explanation of Answer (B) Correct - per 2BwCA-0.3, "It is preferred to prepare 4KV ESF Bus 241 for the Unit 1 crosstie to support the motor driven AF Pump availability. (A) Incorrect - bus selection is ok, but Unit 2 must align Train B loads to support Unit 1 operation. (C&D) Incorrect - 242 is not the preferred crosstie power source if 241 is available.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Response to Opposite Unit Loss of All AC	2BwCA-0.3	step 4 NOTE	4	1WOG	1C

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 104 RO Number: SRO Number: 78

Question Loss of Offsite and Onsite Power (Station Blackout)

While performing steps of 1BwCA-0.0, "Loss of All AC Power", which of the following actions, if performed in the Main Control Room, will NOT result in the desired system/component response?

- a. Actuate Main Steamline Isolation
- b. Reset Containment Isolation Phase A
- c. Close CC from RCPs thermal barrier isol valve, 1CC685
- d. Sync and Close Bus 241/141 reserve feed breaker, ACB 1414

Answer c **Exam Level** R **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

055 Station Blackout

EA2. Ability to determine and interpret the following as they apply to Station Blackout:

EA2.04 Instruments and controls operable with only dc battery power available 3.7 4.1

Explanation of Answer DC power supplies 125VDC for both ESF divisions, including Rx trip switchgear, MCB ESF section, ESF switchgear control systems. MSIVs will close, cnmt isol phase A will reset, ACB 1414 will close. (A) Incorrect - MSIVs will close (B) Incorrect - Phase A will reset (C) Correct-1CC685 is a MOV and has no power for the breaker or motor operator. (D) Incorrect - 1414 will close

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of All AC Power	1BwCA-0.0		2,5	8	
Bwd Big Notes - 125 VDC System	DC-1			3	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 105 **RO Number:** 81 **SRO Number:**

Question Loss of Offsite and Onsite power (Station Blackout)

The following conditions exist on Unit 1:

- A loss of all AC power occurred 20 minutes ago
- The Emergency Director has classified the event in progress as a Site Emergency
- All State and NRC initial notifications have been made as required
- Maintenance now estimates 5 hours to restore AC power to either ESF Bus
- The Emergency Director has upgraded the classification to a General Emergency
- The time now is 01:15

The State of Illinois must be notified of this change in emergency plan classification NO LATER THAN:

- a. 01:30
- b. 02:00
- c. 02:15
- d. 02:30

Answer a Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1
 055 Station Blackout
 2.4 Emergency Procedures / Plan
 2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies. 2.2 3.6

Explanation of Answer Per EP-AA-114 "Notifications" - offsite notifications must be made within 15 minutes of any classification level change. (A) is only Correct time frame.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Notifications	EP-AA-114	4.1.1	1	1	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 106 RO Number: SRO Number: 79

Question Loss of DC Power

The following conditions exit on Unit 1:

- A turbine trip / reactor trip has occurred concurrent with a loss of DC Bus 113
- The crew has completed the Immediate action steps of 1BwEP-0, "Reactor Trip or Safety Injection"
- Transition has been made to 1BwEP ES-0.1, "Reactor Trip Response"
- Concurrently, the SRO has entered 1BwOA ELEC-1, "Loss of DC Bus"

Which of the following describes why an operator is dispatched in 1BwOA ELEC-1 to locally open the PMG breaker?

- a. Prevent reverse rotation of the 1A Reactor Coolant Pump
- b. Half of the steam dump valves have failed open
- c. Protect equipment from low frequency / voltage AC
- d. Half of the feedwater isolation valves have failed open

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 2

058 Loss of DC Power

AK3. Knowledge of the reasons for the following responses as they apply to Loss of DC Power:

AK3.02 Actions contained in EOP for loss of dc power 4.0 4.2

Explanation of Answer (C) Correct - the main generator remains connected to the UAT. 4KV bus 143 and 6.9 bus 157 will have lost breaker control power and cannot ABT. They will remain energized, will all attendant loads, from the main generator as long as the PMG remains closed. As the generator slows, voltages and frequency drops. (A) Incorrect - although 1A RCP remains energized, it will not reverse direction. (B) Incorrect - steam dumps fail closed and are not affected by DC 113. (D) Incorrect - feedwater isolation valves fail closed and are controlled via the ESF DC Busses 111 and 112.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Operation Procedures	1BwOA ELEC-1	Attachment A	3	100	
Bwd Big Notes	DC-1		1	3	
BwOA ELEC-1 Lesson Plan	11-OA-CL-01	II	2	6	3,4

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type Comment

Record Number: 107 RO Number: 82 SRO Number: 80

Question Accidental Gaseous Radwaste Release

Which of the following Aux Building Ventilation System design operating characteristics MINIMIZES the spread of an accidental gaseous radwaste release?

- a. High radiation in the plant vent stack shuts down the VA supply fan
- b. High radiation in the plant vent stack shuts down the VA exhaust fan
- c. Aux Building pressure is maintained positive with respect to surrounding areas
- d. Aux Building pressure is maintained negative with respect to surrounding areas

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

060 Accidental Gaseous Radwaste Release

AK3. Knowledge of the reasons for the following responses as they apply to Accidental Gaseous Radwaste Release:

AK3.02 Isolation of the auxiliary building ventilation 3.3* 3.5*

Explanation of (A&B) Incorrect - there are no automatic actions associated with Aux Building Ventilation due to high rad. (C)

Answer Incorrect - Aux Building is normally maintained at a negative pressure to keep radiation from spreading (D) is Correct

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Bwd Operator Big Notes	Aux Bldg Vent	VA-2		0	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 109 **RO Number:** 84 **SRO Number:** 82

Question Area Radiation Monitoring (ARM) System Alarms

Which of the following function(s) are provided by the Auxiliary Building General Area radiation monitors?

1. Trending of current and past radiological conditions
 2. Local alarms for personnel protection
 3. Detection of unauthorized radioactive materials movement
 4. Automatic start of Aux Building Charcoal Booster fans
- a. 2, 3, 4 ONLY
 - b. 1, 2 ONLY
 - c. 1, 2, 3 AND 4
 - d. 1, 2, 3 ONLY

Answer d **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
061 Area Radiation Monitoring (ARM) System Alarms
2.1 Conduct Of Operations
2.1.27 Knowledge of system purpose and or function. 2.8 2.9

Explanation of Answer Aux Building Charcoal Booster fans auto start upon receipt of a SI signal only. Not from Hi rad. (A-C) are Incorrect (D) is the Correct answer

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Normal Operating Procedures	BwOP VA-5	E	2	10	
Radiation Monitors LP	I1-AR-XL-01 (49)	I.A	1	2	1

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type **Comment**

Record Number: 110 **RO Number:** 85 **SRO Number:** 83

Question Loss of Nuclear Service Water

The following conditions exist on Unit 1

- Power level is 100%, steady state
- 1A SX pump just tripped on overcurrent
- 1B SX pump could NOT be started
- Only 1 SX pump is available on Unit 2
- Conditions on Unit 1 require cross tie of SX systems

What actions are taken to reduce the heat loads on the SX System(s) when cross-tying units with only ONE SX pump available?

- a. ONE CCW heat exchanger on each unit is isolated
- b. ONE RCFC train on each unit is shutdown and isolated
- c. All containment chillers on BOTH units are stopped and isolated
- d. SX flow to all RCFC's on ONE unit is isolated

Answer b Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

062 Loss of Nuclear Service Water

AA1. Ability to operate and / or monitor the following as they apply to Loss of Nuclear Service Water:

AA1.01 Nuclear service water temperature indications 3.1 3.1

Explanation of Answer (A) incorrect - isolation of 1 heat exchanger on each unit invokes LCO 3.0.3. (B) Correct - per 1BWOA PRI-8, attach B step 1b. RNO. (C) Incorrect - renders control of containment temperatures not possible (D) Incorrect - renders RCFCs incapable of controlling cnmt temperatures inoperable on one unit.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Operating Procedure	1BWOA PRI-8	Attach B	16	100	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type Comment

Record Number: 111 RO Number: 86 SRO Number: 84

Question Loss of Instrument Air

The following conditions exist on Unit 1:

- Reactor power is 100%, steady state with all systems in automatic control
- A secondary transient is preceded by the following indications:
- Annunciator 1-21-E10, "125 VDC DIST PNL 111/113 VOLT LOW" alarm LIT
- MCB Indicator 1EI-DC001, "DC BUS 111 VOLTAGE" indicates 0

The IMMEDIATE action required to be taken by the operating crew is to:

- a. Assume local emergency control of safe shutdown equipment
- b. Start-up/restore the 125 VDC ESF Bus Battery Charger
- c. Cross-tie/restore the 125 VDC ESF Bus to Unit 2 ESF DC Power
- d. Verify Unit 1 reactor and turbine are tripped and ESF Busses are energized

Answer d Exam Level S Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 3 SRO Group: 2
 065 Loss of Instrument Air
 2.4 Emergency Procedures / Plan
 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. 4.0 4.0

Explanation of Answer Loss of DC Bus 111 has occurred as evidenced by annunciator 1-21-E6 and DC Bus voltage indicator. Loss of ESF DC results in loss of IA to the main feed regulating valves, which fail closed. Resulting closure of MFRVs results in or required an immediate reactor trip due to potential loss of heat sink. (D) Correct. (A) Incorrect - safe shutdown equipment on Train B is not affected by loss of DC power. Train A equip will not be operated locally w/o tripping protection if an operable train is available. (B) Incorrect - no indications exist that the battery charger has tripped off. (C) Incorrect - While it may be desirable to cross-tie DC Busses at some point, must first determine status of why 111 has tripped. Before that, the immediate concern is the failure of all FRVs on Unit 1, lowering SG level and imminent reactor trip.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Abnormal Operating Proc - Loss of DC Bus	1BwOA ELEC-1	Attach A	3	100	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 112 RO Number: SRO Number: 85

Question Loss of Containment Integrity

Which of the following transients is analyzed to result in the highest containment pressure AND greatest leakage out of containment?

- a. Design basis LOCA
- b. Design basis Steam Line Break inside containment
- c. Inadvertant containment spray actuation
- d. Pressurizer vapor space LOCA

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

069 Loss of Containment Integrity

AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Containment Integrity:

AK1.01 Effect of pressure on leak rate 2.6 3.1

Explanation of Worst case LOCA generates larger mass and energy release than the worst case steam line break.

Answer Inadvertant CS actuation would cause pressure to decrease, even if all RCP seals failed a DB LOCA is a larger mass and energy release.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
FR-Z Containment	11-FR-XL-05	II	2	1	3
Technical Specifications	3.6.4	Basis	B.3.6.4-1	0	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments: 2000 Bwd NRC

Comment Type Comment

Record Number: 113 RO Number: 87 SRO Number: 86

Question Inadequate Core Cooling

Unit 1 reactor is shutdown with RCS pressure at 485 psig and decay heat being removed by the steam generators. In order to avoid approaching an inadequate core cooling situation, what pressure must be maintained in the steam generators to obtain a 50°F subcooling margin in the RCS? (Assume a negligible delta-T exists between the RCS and the steam generators)

- a. 285 psig
- b. 465 psig
- c. 665 psig
- d. 785 psig

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

074 Inadequate Core Cooling

EK1. Knowledge of the operational implications of the following concepts as they apply to Inadequate Core Cooling:

EK1.08 Definition of subcooled liquid 2.8 3.1

Explanation of Answer calculated value with the steam tables for a pressure of 485 psig. (500 psia has Tsat of 467°F. 50°F subcooled is 417°F Psat for 417°F is 300 psia = 285 psig)

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Inadequate Core Cooling LP Steam Tables	I1-IT-XL-01	I	1	1	2

Material Required for Examination Steam Tables

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: using steam tables to determine saturation at 485 psig and 50°F lower - to 285 psig (A)

Comment Type **Comment**

Record Number: 114 **RO Number:** 88 **SRO Number:** 87

Question Inadequate Core Cooling

While operating at Rated Thermal Power, a Large Break LOCA resulting in Containment Spray actuation occurred on Unit 1.

Core exit thermocouples are indicating 800°F and increasing.

Reducing demand on which of the following controllers will result in REDUCING cooling flow to the core?

- a. 1CV-182, Charging Header Backpressure Control Valve.
- b. 1RH-607, RH Heat Exchanger Outlet Flow Control Valve.
- c. 1RH-619, RH Heat Exchanger Bypass Flow Control Valve.
- d. 1RY-455B, Pressurizer Spray Valve.

Answer b **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
074 Inadequate Core Cooling

EK2. Knowledge of the interrelations between Inadequate Core Cooling and the following:

EK2.09 Controllers and Positioners 2.6* 2.6*

Explanation of Answer B. Correct. Decreasing demand on this controller will reduce flow from the RH pump to be injected into the core because it is on the discharge of the pump and normally aligned 100% open. A. Incorrect. decreasing demand on 1CV-182 will not decrease charging flow to the core because this path is isolated. C. Incorrect. This valve is normally fully closed at 100% power. D. Incorrect. closing a spray valve will not decrease flow to the core because there will be no RCPs running to affect RCS pressure.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
System big notes	dwgs CV-1, RH-1			4,3	
Op Action Summary Page	1BwEP-0	Trip RCPs When			

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 115 **RO Number:** 89 **SRO Number:** 88

Question rediagnosis

With Unit 1 operating at 100% power, the following events occurred:

- A reactor trip, coincident with a loss of Instrument Bus 114
- All systems responded as expected after the trip

With NO operator actions, 5 minutes after the trip Steam Generator water levels will be

- a. HIGHER than normal post trip response due to a delay in ISOLATING AFW flow and the Rediagnosis procedure 1BwEP ES-0.0 should be used.
- b. HIGHER than normal post trip response due to a delay in ISOLATING AFW flow and the Rediagnosis procedure 1BwEP ES-0.0 should NOT be used.
- c. LOWER than normal post trip response due to DECREASED AFW flow and the Rediagnosis procedure 1BwEP ES-0.0 should be used.
- d. LOWER than normal post trip response due to DECREASED AFW flow and the Rediagnosis procedure 1BwEP ES-0.0 should NOT be used.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 2 SRO Group: 1
 E01 Rediagnosis

EK2. Knowledge of the interrelations between Rediagnosis and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.5 3.8

Explanation of Answer A loss of inst bus 114 will cause the B train of AFW flow control valves to close after flow is sensed through them. This reduces the total AFW flow to the SG's, reducing post trip level response to just one train of AFW vice 2. Use of rediagnosis is limited to those events where SI has actuated or is required. No SI actuated or is required in this case. A,B incorrect, plausible if AFW flow control valves failed open not closed. C. incorrect - rediagnosis does not apply. D. Correct

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss of Instrument Bus	1BwOA ELEC-2	Table D	18		7A
Rediagnosis	1BwEP ES-0.0	Purpose	1		

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Direct From Source

Question Source Comments: 2001 Bwd NRC

Comment Type Comment

Record Number: 116 RO Number: 90 SRO Number: 89

Question LOCA Outside Containment

While performing 1BwCA-1.2, "LOCA Outside Containment", under what condition would 1SI8835, "SI Pumps to Cold Leg Isolation Valve", remain closed after being repositioned?

- a. RCS pressure is increasing
- b. SI pump discharge pressure is increasing
- c. Pressurizer level is decreasing
- d. CETC temperatures are decreasing

Answer a **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

E04 LOCA Outside Containment

EK3. Knowledge of the reasons for the following responses as they apply to LOCA Outside Containment:

EK3.2 Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment). 3.4 4.0

Explanation of (A) Correct - as stated in the procedure 1BwCA-1.2 as the leak is isolated with an RCS pressure increase.

Answer (B,C,D) are not options as given in the procedure

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
LOCA Outside Containment (CA)	1BwCA-1.2	NOTE	4	1A WOG 1C	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type **Comment**

Record Number: 118 **RO Number:** 91 **SRO Number:** 91

Question Pressurized Thermal Shock

The following conditions exist on Unit 1

- Steam generator 1A tube rupture has occurred
- Crew is performing the initial RCS cooldown step of 1BwEP-3, "Steam Generator Tube Rupture"
- All RCP's are OFF
- Loop 1A Tc indicates 180°F

The STA has reported an ORANGE path on RCS Integrity.

The Unit Supervisor should _____(1)_____ because _____(2)_____:

_____ (1) _____

_____ (2) _____

- | | |
|--|---|
| a. Remain in 1BwEP-3 until the second RCS depressurization is complete | Cold injection water is cooling Loop 1A Tcold |
| b. Immediately transition to 1BwFR-P.1, "Response to Imminent PTS" | A severe challenge exists to the CSF |
| c. Transition to 1BwFR-P.1 as soon as the initial cooldown is complete | Cooldown in 1BwEP-3 takes priority over 1BwFR-P.1 |
| d. Remain in 1BwEP-3 until the appropriate SGTR recovery procedure is selected | An RCP will be started in 1BwEP-3 |

Answer a **Exam Level** S **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E08 Pressurized Thermal Shock

EA2. Ability to determine and interpret the following as they apply to Pressurized Thermal Shock:

EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. 3.5 4.1

Explanation of Answer (A) correct - per 1BwEP-3 Caution prior to step 6 and Note prior to step 28 (B) Incorrect - same caution states NOT to do FR-P.1 at this time. (C) Incorrect - same caution states to wait until completion of step 28 (D) Incorrect - caution say wait until step 28, not end of the procedure

	Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
SGTR		1BwEP-3	step 6, 28	10,36	100WO	G1C

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type **Comment**

Question Pressurized Thermal Shock

Which of the following reflects the intent of the major actions performed in 1BwFR-P.1, "Response to Imminent Pressurized Thermal Shock"?

- a. Reduce RCS cooldown rate and decrease RCS pressure
- b. Reduce RCS cooldown rate and increase RCS pressure
- c. Increase RCS cooldown rate and decrease RCS pressure
- d. Increase RCS cooldown rate and increase RCS pressure

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

E08 Pressurized Thermal Shock

EK3. Knowledge of the reasons for the following responses as they apply to Pressurized Thermal Shock:

EK3.2 Normal, abnormal and emergency operating procedures associated with (Pressurized Thermal Shock). 3.6 4.0

Explanation of Answer (C,D) incorrect - increasing the cooldown rate increases the thermal stresses. (B) incorrect, increasing RCS pressure increases the stresses (A) Correct - reduces thermal and pressure stresses, per 1BwFR P.1

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Response to Imminent PTS	1BwFR-P.1		3-22	1A, WOG 1C	
Response to Imminent PTS	Background Document P-1	1	1	1	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: 2001 Bwd NRC

Comment Type Comment

Record Number: 120 RO Number: 92 SRO Number: 93 Number(s) n

Question Natural Circulation Operations

If operated within Tech Spec limits, which of the following precludes the hot fuel rod in the core from undergoing DNB during a loss of forced coolant flow accident?

- a. Quadrant Power Tilt Ratio
- b. Control Rod Bank Insertion Limit
- c. Nuclear Enthalpy Rise Hot Channel Factor
- d. Control Rod Group Alignment Limit

Answer a Exam Level S Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

E09 Natural Circulation Operations

EA2. Ability to determine and interpret the following as they apply to Natural Circulation Operations:

EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. 3.4 3.8

Explanation of Answer (A) Correct - per TS basis protects against DNB on loss of forced flow. (B,C,D) Incorrect, loss of flow protection against DNB is not included in the safety analysis for any of these limitations

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Tech Specs	Basis	3.1.4,3.1.6,3.2 .2,3.2.4		0,23	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 121 RO Number: SRO Number: 94

Question Natural Circulation Operations

Which of the following describes why it is important to run CRDM fans when performing a natural circulation cooldown?

- a. Provides the heat removal mechanism for the vessel head area
- b. Aids in natural circulation flow through the RCS vessel head region
- c. Prevents erratic indication of SR instruments
- d. Aids in natural circulation flow through the RCS

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 7/17/02
 Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

E09 Natural Circulation Operations

EK2. Knowledge of the interrelations between Natural Circulation Operations and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.6 3.9

Explanation of Answer CRDM fans cool the upper head region that may not be cooled by natural circulation flow. Rx Cavity vent fans provide cooling to the SR NI's.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
EP-0 Series LP	I1-EP-XL-01	VII	38	13	3
Natural Circulation Cooldown	1BwEP ES-0.2	Step 22 RNO	14	WOG1	C

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Direct From Source

Question Source Comments: 1999 Bwd NRC

Comment Type	Comment
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Record Number: 122 RO Number: 93 SRO Number: 95

Question Natural Circulation with Steam Void with/without RVLIS

The following conditions exist on Unit 1

- 1BwEP ES-0.4, "Natural Circulation Cooldown with Steam Void in Vessel (Without RVLIS) Unit 1" is in progress
- RCS Temperature is 450°F
- RCS Pressure is 800 psig
- RVLIS is NOT available
- Charging and letdown flows are matched

With RVLIS NOT available to monitor for void growth in the vessel, which of the following combined indications can be used to verify the presence of a void when letdown flow is increased > charging flow?

RCS pressure will _____(1)_____ and Pressurizer level will _____(2)_____.

_____ (1) _____ (2) _____

- a. Increase Increase
- b. Increase Decrease
- c. Decrease Decrease
- d. Decrease Increase

Answer d Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

E10 Natural Circulation with Steam Void in Vessel with/without RVLIS

EA1. Ability to operate and / or monitor the following as they apply to Natural Circulation with Steam Void in Vessel with/without RVLIS:

EA1.2 Operating behavior characteristics of the facility. 3.6 3.8

Explanation of Answer (D) Correct - rapidly increasing pressurizer level during the RCS depressurization is a sign that voids are forming in the primary system. Pressure decreases, fluid flashes to steam displacing pZR level. (A&B) incorrect - pressure decreases as inventory is removed (C) Incorrect - level increases as voids form

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Nat Circ Cooldown w/o RVLIS	1BwEP ES-0.4	Note - step 8	7	1AWO G1C	
Background Documents	1BwEP ES-0.4	Step 8	33	1C	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 123 RO Number: 94 SRO Number:

Topic

Number(s) **n**

Question Steam Generator Overpressure

The following conditions exist on Unit 1:

- A spurious closure of all MSIVs occurred while operating at 100% power
- The reactor was manually tripped by the operators and immediate actions of 1BwEP-0 were performed
- Recovery operations are in progress utilizing 1BwEP ES-0.1, "Reactor Trip Response"
- The STA has identified a YELLOW path overpressure condition on 1C SG with pressure at 1240 psig
- Checking 1PM04J, there is no steam flow indicated on the 1C steam generator
- All other steam generators and plant safety systems functioned as designed
- During subsequent repairs the unit has been holding in Mode 3 for the past 20 hours

The condition of the 1C steam generator is reportable to the NRC because:

- a. The plant exceeded a safety limit
- b. Challenges occurred to safety valves
- c. A loss of two fission product barriers is imminent
- d. The plant is in a condition prohibited by tech specs

Answer d **Exam Level** S **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3
 E13 Steam Generator Overpressure
 2.4 Emergency Procedures / Plan
 2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies. 2.2 3.6

Explanation of Answer (A) Incorrect - the reportable safety limits are for primary system power level and pressure only. (B) Safety valves are only the Pzr safety valves, not the SG safety valves (C) Incorrect - loss of 2 fission product barriers would result in a Site Emergency which we are not in. Do not meet criteria for potential losses per EALs (D) Correct - with no steam flow and SG pressure at 1240 psig it is above the lift setpoint of all MS safety valve which are inoperable.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Functional Restoration proc SG Overpress	1BwFR H.2		1	1AWO	G1C
FR H.2 background document	FR-H.2	FR-H.2	2-4	1	
Exelon Reportability Manual	LS-AA-1110	SAF 1.11	39		

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 124 **RO Number:** **SRO Number:** 96

Question Steam Generator Overpressure

The following conditions exist on Unit 1:

- An inadvertant FW Isolation occurred on the 1C Steam Generator
- The reactor was manually tripped and 1BwEP-0 "Reactor Trip or SI" performed
- The crew has transitioned to and is performing steps in 1BwEP ES-0.1 "Reactor Trip Response"
- The STA has identified a YELLOW condition on the 1C Steam Generator due to overpressure
- 1C Steam Generator level is indicating off-scale high
- 1C Steam Generator pressure is 1248 psig

Which of the following methods will be available/allowed by performance of 1BwFR-H-2, "Response to Steam Generator High Pressure" to reduce the pressure in the 1C Steam Generator:

- a. Initiate AFW Flow to the 1C Steam Generator
- b. Bleed steam using the Main Steam Dumps
- c. Locally open the 1C PORV
- d. Initiate SG blowdown flow

Answer d **Exam Level** R **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

E13 Steam Generator Overpressure

EK1. Knowledge of the operational implications of the following concepts as they apply to Steam Generator Overpressure:

EK1.2 Normal, abnormal and emergency operating procedures associated with (Steam Generator Overpressure). 3.0 3.3

Explanation of Answer (A) Incorrect - all FW is isolated due to the high level in H.2 and H.3 (B) Incorrect - MSIVs are closed due to the high level to prevent water from entering the steam lines (C) Incorrect - Operators are cautioned NOT to initiate steam release from any steam generator with levels > 93% (D) Correct - After steam cannot be released this is the alternative option per step 8

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Response to SG Overpressure	1BwFR-H.2		1-7	1A	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Record Number: 125 **RO Number:** 95 **SRO Number:**

Question Steam Generator Overpressure

The following sequence of events has occurred on Unit 1:

- Reactor has been manually tripped due to a secondary system malfunction
- 1BwEP-0 has been performed and a transition made to 1BwEP ES-0.1, "Reactor Trip Response"
- The STA has identified a YELLOW path on the Heat Sink Status Tree for steam generator pressure
- The crew has entered 1BwFR-H.2, "Response to Steam Generator Overpressure"
- The crew is preparing to dump steam from the affected steam generator
- The US reads a CAUTION that does not allow releasing steam from a SG with a narrow range level of greater than 93%

Why shouldn't the crew dump steam from the affected SG if NR level is >93%?

- a. May cause an uncontrolled radiation release since it is likely that the steam generator is ruptured
- b. May result in two phase flow and water hammer, potentially damaging pipes and valves
- c. Will be ineffective in lowering SG pressure since the SG water is likely subcooled
- d. Will cause a rapid pressure drop in the RCS, potentially resulting in a safety injection

Answer b **Exam Level** B **Cognitive Level** Memory **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

E13 Steam Generator Overpressure

EK2. Knowledge of the interrelations between Steam Generator Overpressure and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.0 3.2

Explanation of Answer (B) Correct - per FR-H series background documents. (A) Incorrect - no indications are present that would suspect a SGTR had occurred. (C) incorrect - at 1235# and RCS Tave post trip, opening the PORV or Steam Dumps will release steam. (D) Incorrect - no indications to conclude a controlled release cannot be obtained.

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Response to SG Overpressure	1BwFR-H.1		5	1A	
Background Documents	FR-H.2	Caution	12	1C	

Material Required for Examination

Question Source: Other Facility

Question Modification Method: Editorially Modified

Question Source Comments: 2001 Prairie Island NRC

Comment Type **Comment**

Record Number: 126 **RO Number:** 96 **SRO Number:** 97

Question High Containment Pressure

While performing actions of 1BwFR-Z.1, "Response to High Containment Pressure", what steps are taken

to limit the peak pressure rise in containment in the event one of the steam generators is faulted?

- a. All four RCFCs are started in Fast Speed upon entry to 1BwFR-Z.1
- b. Feed Flow is isolated to any steam generator that is depressurizing in an uncontrolled manner
- c. Aux Feedwater Flow to all steam generators is throttled down to 45 gpm per steam generator
- d. All steam generators are allowed to completely depressurize before exiting 1BwFR-Z.1

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 7/17/02

Tier: Emergency and Abnormal Plant Evolutions RO Group: 1 SRO Group: 1

E14 High Containment Pressure

EK2. Knowledge of the interrelations between High Containment Pressure and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.4 3.8

Explanation of Answer (B) Correct - per step 6 of FR-Z.1. (A) incorrect - RCFC's are never run in fast speed in adverse containment conditions to protect the fans (C) incorrect - AFW is only throttled to 45 gpm if all steam generators are faulted. (D) incorrect - all steam generators are not depressurized before exiting Z.1

Reference Title	Facility Reference Number	Section	Page	Revisio L. O.
Response to High Cnmt Pressure	1BwFR-Z.1		9	1AWO G1C

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Record Number: 127 RO Number: 97 SRO Number: 98

Question High Containment Pressure

1BwCA-1.1, "Loss of Emergency Coolant Recirculation", is in progress when an ORANGE path is identified for containment pressure. 1BwFR-Z.1, "Response to High Containment Pressure", is entered immediately and containment isolation is verified. The operators then operate the containment spray system according to the directions found in 1BwCA-1.1, instead of 1BwFR-Z.1.

Under these conditions, 1BwCA-1.1 takes precedence over 1BwFR-Z.1 because the 1BwCA-1.1 pump operating criteria:

- a. Ensure that the maximum heat removal system capacity is used to reduce containment pressure.
- b. Are more restrictive, ensuring continuous containment spray system operation to reduce containment pressure.
- c. Are less restrictive, permitting reduced containment spray operation to conserve RWST water.
- d. Provide a more rapid means of verifying automatic actuation of the containment spray system.

Answer C **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E14 High Containment Pressure

EK3. Knowledge of the reasons for the following responses as they apply to High Containment Pressure:

EK3.2 Normal, abnormal and emergency operating procedures associated with (High Containment Pressure). 3.1 3.7

Explanation of Answer (C) Correct - spray operation requirements are relaxed to allow conservation of RWST water inventory (A) incorrect - CS is not maximized but reduced (B) incorrect - 1.1 criteria is less restrictive, allowing no CS pumps if all RCFCs are available and running in accident mode. (D) incorrect - actuation is not verified until step 9, and then only to look at required flows

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Response to High Containment Pressure	1BwFR-Z.1	Caution	3	1A	
Loss of Emergency Coolant Recirculation	1BwCA-1.1	step 9 - table	8	100	

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type	Comment

Record Number: 128 **RO Number:** 98 **SRO Number:** 99

Question High Containment Radiation

The following conditions exist on Unit 1:

- A small break RCS LOCA occurred 45 minutes ago
- The reactor was successfully tripped and SI actuated
- RCS pressure is 900 psig and increasing slowly
- RCS temperature is 500°F and decreasing slowly
- Pzr level is 25% and increasing slowly
- Containment pressure is 4 psig, decreasing slowly from a peak pressure of 22 psig
- Containment radiation levels are steady at 2.6E5 R/hr
- All S/Gs are intact with NR levels at 27% and increasing slowly
- The operating crew has performed all applicable steps of 1BwEP-0 and have transitioned to 1BwEP-1, "Loss of Reactor or Secondary Coolant"

Which of the following statements is true concerning the current plant conditions?

- a. Total Aux Feed Flow may be throttled back to less than 500 gpm to reduce RCS cooldown effects
- b. Containment Spray pumps may now be stopped at any time deemed appropriate to conserve RWST inventory
- c. RCS subcooling is acceptable and would allow for SI termination if all other parameters are met
- d. Pzr level would require SI be immediately reinitiated if it had been previously terminated

Answer d **Exam Level** B **Cognitive Level** Application **Facility:** Braidwood **ExamDate:** 7/17/02

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

E16 High Containment Radiation

EA1. Ability to operate and / or monitor the following as they apply to High Containment Radiation:

EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. 3.1 3.2

Explanation of Answer Adverse containment conditions exist due to the high rad levels (>1E5 r/hr). (A) incorrect - requires SG levels between 31-50%. (B) incorrect - requires run time of >2 hours. (C) incorrect - 500°F requires 950psig adverse cnmt. (D) Correct - Pzr level is required to be >28%

Reference Title	Facility Reference Number	Section	Page	Revisio	L. O.
Loss Of Reactor or Secondary Coolant EP	1BwEP-1	OAS, steps 3,6,7		100WO	G1C

Material Required for Examination

Figure 1BwEP 1-1 RCS Subcooling Margin

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 130 **RO Number:** 100 **SRO Number:** 100