

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos: 50-456; 50-457
License Nos: DPR-29; DPR-30

Report Nos: 50-456/98305(OL); 50-457/98305(OL)

Licensee: Commonwealth Edison Company (ComEd)

Facility: Braidwood Nuclear Power Station, Units 1 and 2

Location: RR# 1, Box 79
Braceville, IL 60407

Dates: September 14, 1998
Telephoned Examination Results October 20, 1998

Inspectors: H. Peterson, Chief Examiner, RIII

Approved by: M. Leach, Chief, Operator Licensing Branch
Division of Reactor Safety

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EXECUTIVE SUMMARY

Braidwood Nuclear Generating Station
NRC Examination Reports 50-456/98305; 50-457/98305

The Braidwood training department in conjunction with Byron training department developed an initial operator licensing examination that was administered to one Braidwood Reactor Operator license applicant by NRC examiners.

Results:

- The applicant passed the retake examination and was issued a Reactor Operator's license.

Report Summary:

- The training staff's knowledge of the examination development guidelines, attention to detail during examination development, and the ability to develop technically accurate written examination material in accordance with the examination guidelines were considered satisfactory. The licensee performed quality assurance reviews and satisfactorily submitted the written examination for NRC approval. (Section O5.2)
- The inspectors determined the training staff properly administered the written examination. No examination compromise issues were identified and examination security was considered good. (Section O5.3)
- Taking into account the combined written examination results of Braidwood and Byron, the high failure rate and below average grades on the written examination suggested that the training programs did not well prepare the applicant for the examination. Several apparent knowledge deficiencies were identified through the written examination, including some understanding of system response, knowledge of operator actions, and bases for technical specifications. (Section O5.4)
- The examiners accepted four out of the five post written examination comments. Overall, the licensee's submittal of the post examination documents was considered satisfactory. (Section O5.5)

Reports Details

I. Operations

O5 Operator Training and Qualification

O5.1 General Comments - Initial Operator License Examination

An initial licensing examination was administered to one Reactor Operator (RO) applicant in conjunction with the Byron initial license examination at the Byron Station. The written examination was administered by Byron training staff with approval from and observation by the NRC examiners on September 14, 1998.

The licensee developed the initial operator license examination in accordance with guidance prescribed in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8. In general, the examiners reviewed and approved all examination material that the licensee developed prior to its administration.

O5.2 Pre-Examination Activities

a. Examination Scope

The licensee developed the examination material in accordance with the prescribed examination development guidelines. The examiners reviewed, revised, and validated the written examination material during the week of August 31, 1998.

b. Observations and Findings

Written Examination

The written examination was primarily developed by Byron training department with site specific information updated to reflect Braidwood Station. The Braidwood training department staff reviewed and verified the written examination prior to NRC submittal in accordance with NUREG 1021 ES 401-6, "Written Examination Quality Assurance Checklist." The examiners reviewed all 127 questions from the originally submitted written examination. The examiners identified that some question deficiencies that required additional corrections and enhancements to better conform with the written examination question development guidance stated in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8. The licensee was informed of the potential changes and improvements needed on the written examination. During the on-site validation week, additional effort was made to ensure the required changes and enhancements were made to properly reflect the examination guidance to allow for exam administration. Following the validation week, the licensee had one week to incorporate the changes and enhancements to the written examination.

c. Conclusions

The training staff's knowledge of the examination development guidelines, attention to detail during examination development, and the ability to develop technically accurate written examination material in accordance with the examination guidelines were considered satisfactory. The licensee performed quality assurance reviews and satisfactorily submitted the written examination for NRC approval. For specific details on the written examination deficiencies refer to Byron initial examination report 50-454/98301(OL); 50-455-98301(OL).

O5.3 Examination Administration

a. Examination Scope

The written examination was administered on September 14, 1998, by Byron training staff with approval from and observation by the NRC examiners.

b. Observations and Findings

Written Examination

The licensee administered the written examination with the approval from and observation by the NRC. The testing facility was appropriate to assure proper examination security. The licensee's examination proctors appropriately implemented their responsibilities in accordance with the guidance of NUREG 1021, Section ES-402. All appropriate documentation for written examination administration was completed. No examination compromise issues were identified.

c. Conclusions

The inspectors determined the training staff properly administered the written examination. No examination compromise issues were identified and examination security was considered good.

O5.4 License Applicant Performance

a. Examination Scope

An initial licensing written retake examination was administered to one RO applicant.

b. Observations and Findings

Written Examination

The one RO applicant passed the written retake examination. But, initially the RO applicant along with the five Byron initial license applicants all failed the written examination. The examiners, taking into account the licensee's post examination comments, regraded the applicant's written examination and the applicant subsequently

passed the test. Following the examination grading, the licensee submitted a list of questions that were missed by 50% or more of the applicants, and a matrix of potential knowledge weaknesses based on the written examination.

c. Conclusions

Taking into account the combined written examination results of Braidwood and Byron, the high failure rate and below average grades on the written examination suggested that the training programs did not well prepare the applicant for the examination. Several apparent knowledge deficiencies were identified through the written examination, including some understanding of system response, knowledge of operator actions, and bases for technical specifications. For further details on the written examination concerns refer to Byron initial examination report 50-454/98301(OL); 50-455/98301(OL).

O5.5 Post Examination Activities

a. Scope

The examiners reviewed the written examination grading that was performed by the licensee in accordance with Form ES-403-1, "Written Examination Grading Quality Assurance Checklist," contained in NUREG-1021, Interim Revision 8. The examiners also reviewed the post written examination comments submitted by the licensee.

b. Observations and Findings

The post examination submittal included the necessary documentation as required per the guidance of NUREG-1021, ES-501. The licensee submitted an analysis of the written examination results, which was a list of missed questions by 50% or more of the applicants, Braidwood and Byron, who jointly took the written examination.

The examiners also reviewed the licensee's submitted post written examination comments. Four out of five comments were accepted by the examiners, and the written examination was graded accordingly. The licensee's comments and the NRC resolution of the comments are detailed in Enclosure 2, "Post Written Examination Facility Comments and NRC Resolution."

c. Conclusions

The examiners accepted four out of the five post written examination comments. Overall, the licensee's submittal of the post examination documents was considered satisfactory.

V. Management Meetings

X1 Exit Meeting Summary

The examiners conducted an exit meeting with members of licensee management on September 22, 1998, and the licensee was contacted by telephone on October 20, 1998, to inform licensee management of the examination results. The licensee acknowledged the

findings presented and indicated that the materials reviewed were not considered proprietary. In addition, members from the Braidwood training department staff attended the Byron senior management meeting conducted with members of Byron and corporate management at the Region III office on October 21, 1998. The purpose of this meeting was to hear and discuss Byron's post examination root cause evaluation concerning the high failure rate on the written examination.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Cerovac, Operations Training Superintendent
T. Benton, ILT Group Lead
P. Hippley, NGG Exam Developer
M. Brown, Byron Training Instructor - Exam Developer

NRC

C. Phillips, Senior Resident Inspector

INSPECTION PROCEDURES USED

NONE

ITEMS OPENED, CLOSED, AND DISCUSSED

NONE

LIST OF ACRONYMS USED

BwAP	Braidwood Administrative Procedure
BwOA	Braidwood Abnormal Operating Procedure
BwOP	Braidwood Operating Procedure
CFR	Code of Federal Regulations
ComEd	Commonwealth Edison Company
DRS	Division of Reactor Safety
EOP	Emergency Operating Procedure
ES	Examination Standards
ILT	Initial Operator Licensing Training
IP	Inspection Procedure
JPM	Job Performance Measure
K/A	Knowledge and Abilities
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
NRC	Nuclear Regulator Commission
NRR	NRC Office of Nuclear Reactor Regulation
OL	Operator Licensing
PDR	Public Document Room
RO	Reactor Operator
SAT	Systematic Approach to Training
SG	Steam Generator
SM	Shift Manager
TS	Technical Specification

Facility Post Written Examination Comments and NRC Resolution

1. EXAMINATION QUESTION RO #3

LICENSEE COMMENT

The question asked how was a procedure change procedurally conveyed to members of the operating crew. Procedure BwAP 350-2 Rev 6, "Daily Order Book," requires only the Shift Manager (SM) to read and initial the Daily Orders and was responsible to ensure appropriate operating personnel were notified, as necessary. The question distractor (A) was incorrect because a "memo" was not issued to all crew personnel and the SM may not be the person who places the information in the Daily Order Book. Distractor (B) was incorrect because the SM was not necessarily informed by memo and it wasn't proceduralized. Distractor (C) was incorrect because individual operators were briefed by the SM (by procedure). Distractor (D) was also incorrect because the Shift Operation Supervisor does not make an announcement at the shift briefings; it was the SM who does. Therefore, there was no definitive correct answer.

Recommend deleting the question.

NRC RESOLUTION

The comment was accepted and the question was deleted.

Question History

No changes were made to the original question as submitted.

2. EXAMINATION QUESTION RO # 10

LICENSEE COMMENT

The question asked the condition for entry into a fuel handling accident given the plant conditions. The entry condition for distractors (B) and (D) directly apply. No procedural differentiation or hierarchy for a backup indication or report to cause the immediate actions to occur. BWOA Refuel-1, Rev. 54, symptoms/entry conditions include: observed dropping of, or damage to, a fuel assembly; observed dropping of an object or a fuel assembly; observed from a potentially damaged fuel assembly. Any of the procedure entry conditions necessitate use of BWOA Refuel-1. Step 3 only requires one of the alarms (AR011 or AR012) in order to execute actions. Expected, conservative operator actions would be to initiate or actuate Unit 1 Containment evacuation alarm when RE-AR012 alarms, followed by a report from personnel in containment or corresponding rise on a duplicate monitor.

Recommend changing the answer key to accept both (B) and (D) as correct answers.

NRC RESOLUTION

The comment was accepted, and both (B) and (D) was noted as correct answers.

Question History

No changes were made to the original question as submitted.

3. EXAMINATION QUESTION RO # 13
LICENSEE COMMENT

The reference cited for this question/answer was BwAP 380-2, Rev. 6E1. This was not valid for the plant conditions stated in the question. This procedure applies for alarms that are in alarm condition >1 shift and for power levels > P-8 set point (30%). The only applicable standard that applies was NOD OPS DEPT STD-BWD which states that "the SER shall be utilized to verify the exact cause of each annunciator..." and "announcement of repetitive alarms ... are not required with the concurrence of the Unit Supervisor..." BwAP 300-1, Rev. 19E1, also applies, giving generic good operating practice guidance to "believe their instruments until the indications are proven to be false." OPS DEPT STDS and BwAP 300-1 are governing guidance.

Recommend changing the answer key to accept both (A) and (C) as correct answers.

NRC RESOLUTION

The comment was accepted, and both (A) and (C) was noted as correct answers.

Question History:

No changes were made to the original question as submitted.

4. EXAMINATION QUESTION SRO # 54
LICENSEE COMMENT

Candidate answered question correctly for current Braidwood plant configuration and procedures in effect. Automatic actuations in combination with designated SG levels would have resulted in both auxiliary feedwater pumps running. The student was advised to use current plant configuration, procedure references and setpoints for the purposes of this examination ("freeze" point of training and evaluation process). This was clarified to him by the ILT group lead in preparation for this examination. The proctor at the point of administering the examination directed use of "NEW" SG numbers (setpoints, data) based on the belief that the student had received this specific training (which he had not). The student had received SG RO training which characterized the modification/purpose, not the setpoints/procedure specifics. Candidate had not attended licensed requal training which reviewed abnormal and emergency operating procedure changes relative to new set points. He was in formal ILT remediation at this point. These changes are not in place yet. He will receive this training during September and October 1998 as part of preparation for license activation.

Recommend changing the answer key to accept choice (A) as the correct answer.

NRC RESOLUTION

The comment was NOT accepted, and the correct answer remains as (B). The original examination question, with the associated answer key and reference based on the New SG information, was submitted to the NRC. The exam documentation with the NRC Quality Assurance checklist (ES 401-6), original submittal and subsequent revision, was accepted and signed by the Braidwood Training Department as part of the NRC exam submittal. The licensee had several opportunities to review, verify, and correct the written examination prior to final submittal to the NRC for exam administration.

Additionally, the licensee's exam development group specifically noted that the examination referenced the new SG and approved by the Braidwood facility.

Question History

Minor wording change on the question stem was made. Changed some information in the question stem that potentially gave information for a previous question. Also, the deleted information was not needed to answer the question. Deleted the word "ONLY" from two distractors, to eliminate "specific determiners" per NUREG 1021, Appendix B. Remainder of the question stem, distractors, and answer were left the same.

5. EXAMINATION QUESTION RO # 96

LICENSEE COMMENT

The question stated that RCS activity was increasing due to corrosion product activation and to identify the effects of placing the cation bed demineralizer in service.

Procedural guidance discriminates the purposes of each ion bed. The cation bed was efficient at removing Lithium which contributes to low pH and further corrosion products and associated activity. BWOA PRI-4 (High Reactor Coolant Activity) provides guidance/direction for action to be taken in the event of high RCS activity. It directs that the standby mixed bed demineralizer be placed in service NOT the cation demineralizer. BWOP CV-8 references use of the Cation Demin for the reduction of pH or fission products, (not for the purpose of reduction of activated corrosion products). Based on procedural guidance and use, the cation demineralizer would not be placed in service, substantiating (A) as a correct answer; (B) would be correct based on the physical properties of the demineralizer, not accounting for procedural guidance and use.

Recommend changing the answer key to accept both (A) and (B) as correct answers.

NRC RESOLUTION

The comment was accepted, and both (A) and (B) was noted as correct answers.

Question History

No changes were made to the original question as submitted.

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

Applicant Information

Name: MASTER EXAMINATION	Region: III
Date: SEPTEMBER 14, 1998	Facility/Unit: Braidwood 1 & 2
License Level: RO	Reactor Type: W
Start Time:	Finish Time:

Instructions

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

Applicant Certification

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

Applicant's Signature

Results

Examination Value	99 100 Points
Applicant's Score	_____ Points
Applicant's Grade	_____ Percent

SUBJECTIVE SCORE INSTRUCTOR USE ONLY				
100	90	80	70	60
50	40	30	20	10
9	8	7	6	5
4	3	2	1	0

IMPORTANT

TO USE SUBJECTIVE SCORE FEATURE

• MAKE DARK MARKS
• ERASE COMPLETELY TO CHANGE
• EXAMPLE: 1 2 3 4 5 6 7 8 9 10

• MARK TOTAL POSSIBLE SUBJECTIVE POINTS
• ONLY ONE MARK PER LINE ON KEY
• 150 POINTS MAXIMUM

EXAMPLE OF STUDENT SCORE: 1 2 3 4 5 6 7 8 9 10

TEST RECORD	
PART 1	
PART 2	
TOTAL	

NAME		TEST NO.	
SUBJECT		HOUR	
DATE			

PART 1

1998 INITIAL LICENSE TRAINING

Description: NRC REACTOR OPERATOR FINAL EXAMINATION

① 99.0 B 9/2/98

Total Points: 100.0 Points Received: _____

NRC APPROVED
BRAIWOOD KEY

Name/Date: _____

Instructions

1. Use black ink only on all portions of the exam package EXCEPT for the scantron answer selections.
2. Print your name and date in the space provided above.
3. If you have any questions or need clarification during the examination, notify the proctor.
4. Conversation during the examination is prohibited except when asking for question clarification from the proctor.
5. Cheating on the examination will result in failure of the examination and may result in further disciplinary action.
6. Use only #2 pencil to mark your selection on this exam sheet.
7. Completely darken the selected answer. If you make a mistake, completely erase the darkened selection.
8. Ensure you do not skip a question or answer which would place you out of sequence.
9. Do not place any extraneous marks on this exam sheet.
10. You have 4.0 hour(s) to complete this exam.
11. Prior to handing in your exam, verify that you have transferred your answers to this scantron sheet properly.

I have neither given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam's integrity. I also understand my obligation to report any exam compromise by others prior, during, or subsequent to the exam administration.

signature _____

date / /

(T)	(F)	KEY			
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Deleted
NRC accepted

(B/D) NRC accepted

(A/B) NRC accepted

50

① See attached justification

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FEED THIS DIRECTION

FORM # 999-12 11 10 9 8 7 6 5 4 3 2 1

SUBJECTIVE SCORE INSTRUCTOR USE ONLY

400	300	200	100	
350	250	150	50	
300	200	100	50	
250	150	100	50	

(T) (F) KEY

IMPORTANT

TO USE PROACTIVELY

Check your results by entering your score and name on the back of this answer sheet.

TO OBTAIN RESULTS

1. NAME

2. SUBJECT

3. TEST NO.

4. DATE

5. SCORE

PART 2

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NR C
NOT accepted w/ answer stands as (B)

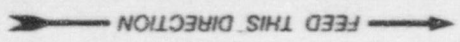
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NR C
NOT accepted w/ answer stands as (B)

① See attached justification



Question / Evaluation of requirement for "active" license

An "Active" licensed NSO (original license obtained in 1996) worked the following schedule at Braidwood:

- 9/4 - 0700 to 1500 as Unit 1 NSO
- 9/7 - 0700 to 1500 as Unit 2 NSO
- 9/8 - 0700 to 1500 as Unit 2 NSO
- 9/9 - 0700 to 1200 as Unit 2 NSO and 1200 to 1500 as WEC NSO
- 9/10 - 0700 to 1500 as WEC NSO
- 9/11 - 0700 to 1500 as Unit 1 NSO
- 9/14 - 1500 to 2200 as Unit 2 NSO
- 9/12 - 1500 to 2200 as Unit 2 NSO

The NSO...

- a. meets the requirements for maintaining his/her license active for the next quarter.
- b. needs to work an additional FOUR hour shift to maintain his/her license active for the next quarter.
- c. needs to work an additional EIGHT hour shift to maintain his/her license active for the next quarter.
- d. needs to work TWO additional EIGHT hour shifts to maintain his/her license active for the next quarter.

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 2.1.1 RO Value: 3.7 SRO Value: 3.8 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA Knowledge of conduct of operations requirements.

Explanation of Answer

Reference Title/Facility Reference Number

Revisio L. O.

Braidwood Ops Memo #2-97 issued 5/1/97 rev. 0

Bwd Tsk List

Task P1-AM-TK-180

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 3 Operating Daily Orders

How is a procedure change, which significantly changes normal processes, procedurally conveyed to licensed members of the operating crew?

- a. The SM places the applicable information in the Daily Order Book, and issues an additional memo to all crew personnel that is initialed.
- b. The SM is informed by memo of the addition to the Daily Order Book, and makes an announcement of the addition during the shift briefing.
- c. The SOS places the applicable information in the Daily Order Book, and the individual operator is responsible for reviewing the Daily Order.
- d. The SOS places the applicable information in the Daily Order Book, and makes an announcement of the addition during the shift briefing.

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 2.1.2 RO Value: 3.0 SRO Value: 4.0 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA

Knowledge of operator responsibilities during all modes of plant operation.

Explanation of Answer

Reference Title/Facility Reference Number

Section

Page

Revisio

L. O.

BwAP 350-2 rev. 6

C.7.B.4

14

Intro to Main Control Room Ops Lesson Plan

5

Braidwood Task List

Task P1-AM-TK-025

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question **4** Procedure required usage

An example of a licensed operator evolution that can be performed WITHOUT having a procedure in hand is ...

- a. Adjusting rod position following a boration for delta-I control.
- b. Starting the 1A Heater Drain Pump.
- c. Placing excess letdown in service.
- d. Latching and rolling up the main turbine following surveillance trip test.

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 2.1.23 RO Value: 3.9 SRO Value: 4.0 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Explanation of Answer

Reference Title/Facility Reference Number	Section	Page	Revisio L. O.
Use Of Procedures For Operating Department BwAP 340-1 Braidwood Task List	C.1.f.3)	pg 4,5	rev.12 Task P1-AM-TK-022

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 5 Use of electrical prints

Assuming an auto-close signal is continuously present in the circuit for the 1A SI pump, which contact will be maintained open in order to prevent the starting relay (SR) from attempting repeated breaker closures onto a faulted bus?

(E 1-4030-SI01 is provided for use.)

- a. LC SW
- b. 52/b
- c. Y
- d. LS

Answer C Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 2.1.24 RO Value: 2.8 SRO Value: 3.1 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA

Ability to obtain and interpret station electrical and mechanical drawings.

Explanation of "Y" is an antipump relay that when prevented from energizing interrupts the circuit that energizes the START

Answer relay in the AUTO mstart circuit

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Schematic Diagram Safety Injection Pump 1A 20E-1-4030SI01			
Print Reading Lesson Plan Chap 3	pg 23	rev. 5	2c, 3

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: Braidwood requal bank

Comment Type Comment

Question **6** MOV tagout

An operator is preparing an OOS that designates 1CC685, RCP Thermal Barrier CG Return CNMT Isolation valve, as an isolation point.

What is the acceptability of using this isolation point?

The OOS is...

- a. acceptable if the MOV is tagged at its control switch, power supply and valve handwheel.
- b. acceptable if the MOV is tagged at its control switch, power supply and a blocking device is placed on the valve.
- c. NOT acceptable because the MOV fails to meet isolation requirements.
- d. NOT acceptable because the valve fails open on a loss of power.

Answer	a	Exam Level	B	Cognitive Level	Comprehension	Facility:	Braidwood	ExamDate:	9/14/98
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KA:	2.2.13	RO Value:	3.6	SRO Value:	3.8	Section:	PWG	RO Group:	1	SRO Group:	1
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System/Evolution

KA Knowledge of tagging and clearance procedures.

Explanation of Answer Valve is MOV and requirements include tagging control switch, electrical power supply and local handwheel if accessible.

Reference Title/Facility	Reference Number	Section/Page	Revisio	L. O.
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BwAP 330-1 Out of Service Process		D.4.a pg 12		
		D.4.c.1) pg 14		

F Braidwood Task List	Task P1-AM-TK-010
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Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Question 7 RCS level discrepancy during refueling

The following conditions exist for Unit 1 in preparation for head removal:

- Unit shutdown and cooldown initiated 120 hours ago
- Lowering of RCS level to the reactor vessel flange is underway
- RCS temperature - 95°
- RCS level Control Room indicators: 1LI-RY046 - 401' 0"
1LI-RY049 - 402' 1"
- RH loop 1A in operation with "normal" indications

What is the appropriate action for these conditions?

- a. The lowering of RCS level can continue after verifying appropriate amount of water removed.
- b. The level change must be stopped until the cause for the level discrepancy is determined.
- c. The running RHR pump shall be immediately stopped to prevent cavitation.
- d. The available SI Pump is immediately aligned for hot leg injection and shall be started.

Answer: b Exam Level: B Cognitive Level: Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 2.2.26 RO Value: 2.5 SRO Value: 3.7 Section: PWG RC Group: 1 SRO Group: 1
 System/Evolution

KA Knowledge of refueling administrative requirements.

Explanation of Answer: With any level discrepancy, the reason for the discrepancy must be determined before further draining can continue.

Lesson Title/Facility Reference Number	Section/Page	Revision	L. O.
BwOP RC-4 Reactor Coolant System Drain	D.1	12E1	
BwGP 100-6 Refueling Outage lesson plan		12	2

Material Required for Examination
 Question Source: Facility Exam Bank Question Modification Method: Significantly Modified

Question Source Comments: Zion exam bank

Comment Type Comment
 NRC Significant Industry Event -

Question 8 RO duties in Control Room during refueling

What is a responsibility of the NSO during refueling operations in the main control room?

- a. Checking source range counts while a fuel assembly is being placed in the core.
- b. Verifying direct phone communication with the Fuel Handling Supervisor once per day during fuel movement.
- c. Maintaining a 1/M plot while reloading fuel during a core shuffle.
- d. Updating the Control Room tag board per the Nuclear Component Transfer List on an hourly basis.

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 2.2.32 RO Value: 3.5 SRO Value: 3.3 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA

Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O
BwAP 2000-38 Reactivity Management	F.2.h.5) pg 11	2E2	
Braidwood Task List			Task P1-QG-TK-051

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 9 Radiation exposure determination

An operator has the following exposure history this year until today:

Deep Dose Equivalent (DDE)	-	210 mrem
Committed Effective Dose Equivalent (CEDE)	-	45 mrem
Shallow Dose Equivalent (SDE)	-	33 mrem
Committed Dose Equivalent (CDE)	-	28 mrem

Today the operator was required to make two entries into containment:

- Entry 1: Gamma dose - 52 mrem; Neutron dose - 24 mrem
- Entry 2: Gamma dose - 124 mrem

How much radiation exposure is available to the operator if he has to make additional entries?

His available margin based on the routine Administrative Exposure Control Levels is...

- a. 100 mrem for that day; 2484 mrem for the year.
- b. 100 mrem for that day; 2545 mrem for the year.
- c. 124 mrem for that day; 2569 mrem for the year.
- d. 124 mrem for that day; 2614 mrem for the year.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 2.3.1 RO Value: 2.6 SRO Value: 3.0 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

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

Explanation of Answer Limits are 300 mrem routine DDE/Day and 3000 mrem routine cumulative TEDE/year. C. Neutron rad not counted for daily & yearly; A. All counted for yearly; d. previous DDE+CEDE only counted for year.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Selected BWRPs Lesson Plan		Rev. 00	2,3,4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 10 Fuel Handling Accident Response

The following conditions exist on Unit 1:

- Refueling operations in progress
- A HIGH alarm received on radiation monitor 1RE-AR012, Containment Fuel Handling Incident

When should the NSO initiate action and what action should he/she take from the control room?

Indication of a fuel handling accident is considered when a...

- a. report is received from personnel in containment. The operator starts the containment charcoal filter fans.
- b. report is received from personnel in containment. The operator actuates Unit 1 CNMT evacuation alarm.
- c. corresponding rise is indicated on monitor 1RE-AR011. The operator starts the containment charcoal filter fans.
- d. corresponding rise is indicated on monitor 1RE-AR011. The operator actuates Unit 1 CNMT evacuation alarm.

Answer d Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 2.3.10 RO Value: 2.9 SRO Value: 3.3 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA

Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Explanation of
Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
BWOA REF-1 Lesson Plan		Rev. 0	2,3,4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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The following conditions exist on Unit 1:

- A reactor trip has occurred and both reactor trip breakers are verified open
- The turbine has tripped
- BwEP-0 "Reactor Trip OR Safety Injection" has been entered.
- BUS 141 ALIVE light is NOT lit with bus voltage at ZERO volts
- BUS 142 ALIVE light is lit with bus voltage at 4149 volts.

Which of the following describes the action(s) the operators is/are required to take?

- a. Check SI status.
- b. Turn on the synchroscope and manually close ACB 1412, SAT 142-1 feed breaker.
- c. Manually start 1A D/G and verify ACB 1413, D/G output breaker, closes.
- d. Initiate actions of BwOA ELEC-3 and then check SI status.

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 2.4.16 RO Value: 3.0 SRO Value: 4.0 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA Knowledge of EOP implementation hierarchy and coordination with other support procedures.

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Reactor Trip or Safety Injection BwEP-0	Step 3.b. RNO		
P 2-0 Rx Trip or Si Lesson Plan		rev.11	1,3

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Following transition to BEP-1 "Loss of Reactor Or Secondary Coolant", the US refers to the Operator Action Summary, and directs the operator to Cold Leg Recirculation Switchover Criterion. Which of the following describes the complete set of procedures for which the Transfer to Cold Leg Recirculation requirements are applicable?

(NOTE: The following procedures are in the E-1or CA-1 series:

- BwEP-1 "Loss Of Reactor Or Secondary Coolant"
- BwEP ES-1.1 "SI Termination"
- BwEP ES-1.2 "Post-LOCA Cooldown And Depressurization"
- BwEP ES-1.3 "Transfer To Cold Leg Recirculation"
- BwEP ES-1.4 "Transfer To Hot Leg Recirculation"
- BwCA-1.1 "Loss Of Emergency Coolant Recirculation"
- BwCA-1.2 "LOCA Outside Containment")

- a. BwEP-1, BwCA-1.1 and BwCA-1.2 procedures.
- b. BwEP-1, BwEP ES-1.1 and ES-1.2 procedures.
- c. BwEP-1 and BwEP ES-1.2 procedures.
- d. BwEP-1 procedure.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 2.4.20 RO Value: 3.3 SRO Value: 4.0 Section: PWG IRO Group: 1 SRO Group: 1

System/Evolution

KA Knowledge of operational implications of EOP warnings, cautions, and notes.

Explanation of answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
BwEP-1 Loss of Reactor or Secondary Coolant Lesson Plan		rev.11	1,10

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 13 Identification of inoperable CR annunciators

The following conditions exist on Unit 1:

- Reactor trip breakers status - OPEN
- RCS Tave - 557°F
- Pzr pressure - 2235 psig

Annunciator RCFC VIBRATION HI (1-3-C5) has been in alarm for the past hour due to a vibration condition while maintenance troubleshoots the vibration probe on RCFC 1C.

Which of the following actions is appropriate for this alarm window?

- a. The alarm should be acknowledged for each actuation and the SER monitored for valid alarm inputs.
- b. The alarm should be acknowledged for each actuation and operators stationed locally at each RCFC to monitor vibration.
- c. The alarm should have been silenced without acknowledgement after obtaining Unit Operating Engineer's permission and the SER monitored for valid alarm inputs.
- d. The alarm should have been silenced without acknowledgement with US permission and operators stationed locally at each RCFC to monitor vibration.

Answer C Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 2.4.31 RO Value: 3.3 SRO Value: 3.4 Section: PWG RO Group: 1 SRO Group: 1

System/Evolution

KA

Knowledge of annunciators alarms and indications, and use of the response instructions.

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
RCFC VIBRATION HI /BwAR 1-3-C5	E.	1	51
HANDLING OF MAIN CONTROL BOARD and RADWASTE PANEL ANNUNCIATOR ALARMS/ BwAP 380-2	C.3 C.4		
Braidwood Task List			Task P1-AM-TK-033

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 14 Effect of Xenon Transient & compensation

A feed pump trip occurred resulting in a rapid power reduction on Unit 1. Power was reduced from 100% steady-state conditions using a combination of rods and boration.

The following conditions exist for Unit 1 following stabilization:

- Reactor Power - 60%
- Delta-I target value - +2.0
- Control Bank D position - 160 steps withdrawn
- Tave - 572°F
- Delta-I - -10.5%
- Core Age - MOL

What actions will be required to maintain the current power level and maintain Delta-I within its normal operating band over the next FIVE hours?

- a. Boration and control rod withdrawal, followed by dilution.
- b. Boration and control rod insertion, followed by dilution.
- c. Dilution and control rod withdrawal, followed by boration
- d. Dilution and control rod insertion, followed by boration.

Answer a	Exam Level B	Cognitive Level Application	Facility: Braidwood	ExamDate:	9/14/98
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KA: 001 A2.06	RO Value: 3.4	SRO Value: 3.7	Section: SYS	RO Group: 1	SRO Group: 1
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System: Evolution Control Rod Drive System

KA Ability to (a) predict the impacts of the following on the Control Rod Drive System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Effects of transient xenon on reactivity

Explanation of Answer With delt-I near the negative limit of the band, boration would be initiated to to allow rod withdrawal and hence shifting of power production toward positive delta-I (power shift toward top of core). Later as Xenon (neutron poison) builds in, dilution will be initiated to maintain power level

Reference Title/Facility Reference Number	Section/Page	Revisio L. O.
DELTA I CONSIDERATIONS BwGP 100-8	F.3,5,6	3,4-7

BwGP 100-8 Lesson Plan	rev 4	1
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Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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A problem with the rod control system requires checking several rod bank circuits. The affected power cabinet repairs are to be made by supplying power from the DC hold supply cabinet.

Which statement describes the proper operation for DC Hold and the associated response in the event of a reactor trip?

- a. ONE control rod bank group can be placed on DC HOLD, and these rods will drop if the controls are taken to OFF at the DC Hold cabinet.
- b. ONE control rod bank group and ONE shutdown bank group can be placed on DC HOLD, and these rods will drop if the controls are taken to OFF at the DC Hold cabinet.
- c. ONE control rod bank group can be placed on DC HOLD, and these rods will automatically drop.
- d. ONE control rod bank group and ONE shutdown bank group can be placed on DC HOLD, and these rods will automatically drop.

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 001 K1.03 RO Value: 3.4 SRO Value: 3.6 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Control Rod Drive System

KA Knowledge of the physical connections and/or cause-effect relationships between Control Rod Drive System and the following: CRDM

Explanation of Answer Only one GROUP of control rods can be placed on HOLD at a time in order to ensure the rods are held without falling. Opening the reactor trip breakers interrupts power to the power cabinet and DC Hold cabinet, so that power to the CRDM is interrupted when the breakers open

Reference Title/Facility	Reference Number	Section/Page	Revision	L. O.
Control System	Chap 28	A.5.e pg 40	12	1,9

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 16 Relationship of levels during refueling operations

The following conditions exist for Unit 1:

- Mode 5
- RCS is draining to Pzr level of 40%
- IM calibrations have been completed for LT-048, Refuel Cavity level, in preparation for further draining
- LI-462 indicates 40%

What is the relationship of Pzr level instrument LI-459 as compared to LI-048?

- a. LI-459 and LI-RY048 will be offscale high.
- b. LI-RY048 will be just onscale and LI-459 will be offscale low.
- c. LI-459 will read higher than LI-462 and LI-RY048 will just be onscale.
- d. LI-RY048 will be offscale high and LI-459 will read lower than LI-462.

Answer C Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 002 A1.11 RO Value: 2.7 SRO Value: 3.2 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Reactor Coolant System

KA Ability to predict and/or monitor changes in parameters associated with operating the Reactor Coolant System controls including: Relative level indications in the RWST, the refueling cavity, the PZR and the reactor vessel during preparation for refueling

Explanation of Answer LI-462 is the cold calibrated Pzr level instrument and will read lower (but more accurately) than the hot calibrated level instruments (LI-459/460/461) at lower RCS temperatures. The refueling cavity level instrument just comes onscale at 40% Pzr level.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
REACTOR COOLANT SYSTEM DRAIN			
BwOP RC-4	D.2 pg 4	rev. 12E1	
BwOP RC-4A5			
BwCB 1/2 fig 31			
BwGP 100-6 Refuel Outage lesson plan		rev. 12	1,2

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist for Unit 1:

- Reactor power - 100%
- RCS activity is elevated, but below Technical Specification (CTS) levels
- Pzr pressure - 2225 psig
- Pzr level - 44%
- Leak rate - 10 gpm

In an attempt to isolate the leakage past the PORV, the Block Valve 1RY8000B was taken to close. The Block Valve failed to close and the operator placed 1RY456 in the CLOSE position. When conditions stabilize:

- Reactor power - 100%
- Pzr pressure - 2228 psig
- Pzr level - 44%

How would the operator be able to tell if the PORV has closed?

- a. Position lights for PCV-456 showing CLOSE indication.
- b. Verify stable VCT level indication.
- c. Level change in RCDT.
- d. Lower readings for containment radiation monitors RE-0011A/0012A.

Answer b Exam Level R Cognitive Level Comprehension Facility: Braidwood Exam Date: 9/14/98
 : 002 A3.01 RO Value: 3.7 SRO Value: 3.9 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Reactor Coolant System

KA Ability to monitor automatic operations of the Reactor Coolant System including: Reactor coolant leak detection system

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
1BwAR 12-C-6		rev51E2	
Braidwood Task List			task P1-OA-TK-058

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Question 18 Use of Loop Isolation Valves

The following conditions exist on Unit 1:

- RCS Loop C is isolated for maintenance
- RCS Loop A had been isolated for maintenance
- RCS Loop A Hot Leg Stop Isolation Valve (LSIV) was opened at 1001
- RCS Loop A Bypass Stop Valve was opened at 1005 with relief line flow of 115 gpm verified
- RCS Loop A Cold Leg LSIV is closed
- RCS temperature - 110°F
- RCS Hot Leg Loop temperatures - 108°F (A); 119°F (B); 110°F (C); 125°F (D)
- RCS Cold Leg Loop temperatures - 103°F (A); 108°F (B); 90°F (C); 115°F (D)
- S/G levels (Narrow Range) - 20% (A); 30% (B); 15% (C); 32% (D)

What will occur when the operator takes the control switch for MOV-RC8002A (RCS Loop A Cold Leg LSIV) to OPEN at 1509?

The valve...

- a. will travel fully open with NO automatic actuations.
- b. will travel fully open, and the AFW pumps get a start signal.
- c. remains closed because the temperature difference interlock remains active.
- d. remains closed because the timer interlock is still active.

Answer a Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
: 002 K4.09 RO Value: 3.2 SRO Value: 3.2 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Reactor Coolant System

KA Knowledge of Reactor Coolant System design feature(s) and or interlock(s) which provide for the following:
Operation of loop isolation valves.

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Simplified RCS/RC-1	valve interlocks/1	3	
Reactor Coolant system lesson plan			
Chapter 12		8	9

Material Required for Examination

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

Question Source Comments: Question 30/35 on Braidwood 1996 NRC exam is about LSIV interlocks. Premise and answers significantly different. Question asked about interlock for opening HL LSIV.

Comment Type Comment

Question 19 RCP and Pzr spray operations

The following Unit 1 conditions exist:

- RCS temperature (Average CETC) - 140°F
- RCS pressure - 365 psig
- A bubble has just been drawn in the Pressurizer
- All loops are filled and vented
- Preparations are in progress to start the first RCP for continuous run
- 1C RCP is started

What is the effect on RCS pressure control?

RCS pressure will increase and...

- a. both Pzr Sprays will function normally for Pzr pressure control.
- b. manual cycling of the Pzr heaters will be required for Pzr pressure control.
- c. PORV RY456 will open on high pressure from high pressure bistable PB456E.
- d. Pzr spray will deliver minimal spray flow for Pzr pressure control.

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 003 A1.06 RO Value: 2.9 SRO Value: 3.1 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Reactor Coolant Pump System

KA Ability to predict and/or monitor changes in parameters associated with operating the Reactor Coolant Pump System controls
PZR spray flow

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
BwGP 100-1 Plant Heat up	f. 57 pg 20	rev 11	
BwGP 100-1 Plant Heat up Lesson plan		12	1,2,3

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Question 20 RCP Breaker & interlocks

The following conditions exist on Unit 1:

- Reactor power 26%
- Pzr pressure - 2235 psig
- Pzr level - 35%

RCP 1A breaker trips due to sensed undervoltage from bus 157. What is expected as a result of the trip of the RCP?

- The reactor will trip due to the open RCP breaker.
- The reactor will trip due to RCS loop low flow condition.
- The reactor will be manually tripped by the operator.
- A normal plant shutdown will be initiated.

Answer C Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 003 K2.01 RO Value: 3.1 SRO Value: 3.1 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Reactor Coolant Pump System

KA Knowledge of electrical power supplies to the following:
RCPS

Explanation of Answer No AUTO trip is expected due to power < P-8. Administrative direction for a RCP trip in these conditions is a manual trip will be initiated.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Chp 13, Reactor Coolant Pump lesson plan	C. 4.a 2)/ pg 16	9	8
AC Electrical Distribution lesson plan chp 4		8	10b
C - Memo/ special Op Order			

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 2/ Charging & letdown flows (including seal injection)

The following conditions exist on Unit 1:

- Reactor power - 100%
- PZR pressure - 2235 psig
- PZR level - 44% stable
- CV121 - In MANUAL
- CVCS letdown - Isolated due to leak in Letdown Hx
- CVCS Excess Letdown - In service with maximum flow of 20 gpm
- RCP seal injection - 1A CV pump aligned to all RCPs
- RCP seal leakoff flow - 3 gpm (1A); 3.5 gpm (1B); 3 gpm (1C); 2.5 gpm (1D)

What flow is indicated on Charging Header Flow indicator, FI-121?

- a. 20 gpm
- b. 32 gpm
- c. 55 gpm
- d. 67 gpm

Answer b Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98
 KA: 004 A3.11 RO Value: 3.6 SRO Value: 3.4 Section: SYS RO Group: 1 SRO Group: 1
 System/Evolution Chemical and Volume Control System
 KA Ability to monitor automatic operations of the Chemical and Volume Control System including:
 Charging/letdown
 Explanation of FI-121 indicates total charging flow (chg header + RCP seal flow, less Chg pump recirc (60 gpm)). Flow
 Answer balance - Letdown: $20 + 12 = 32$ & Chg: $0 + 20 + 12 = 32$.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
CVCS/ Schematic	CV-1		
Chp 15a Chemical VolumeControl System lesson plan		10	4,5,9,15

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Question 22 Calculation of dilution

The following conditions exist on Unit 2:

- Unit is in MODE 5
- Unit burnup is 5700 EFPH in Cycle 7
- SDM - 1.3% DeltaK/K
- RCS pressure - 400 psig
- RCS average temperature - 195°F
- RCS boron concentration - 1006 ppm
- Differential boron worth - -10.75 pcm/ppm
- PZR level - 32.3%
- SR NIS count rate - 10 cps , BOTH channels stable background levels
- An inadvertent dilution at 70 gpm begins at 1300 hours

Assuming NO operator action is taken and PZR level remains constant over the time period, when would the HIGH FLUX AT SHUTDOWN alarm actuate?

- a. No action, because BDPS will actuate prior to actuation.
- b. 1430 hours.
- c. 1505 hours.
- d. 1734 hours.

Answer C Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98
 KA: 004 A4.07 RO Value: 3.9 SRO Value: 3.7 Section: SYS RO Group: 1 SRO Group: 1
 System/Evolution Chemical and Volume Control System

Ability to manually operate and/or monitor in the control room:
Boration/dilution

Explanation of Answer Dilution rate $dc/dt = (500)(C)(Y)/M$ where M is the RCS mass at the given temperature (200°F). M = 745,537 lbm; C = 1006 ppm (given); Y=70 gpm (given). The dil rate = 47.2 ppm/hr. HIGH FLUX AT SHUTDOWN alarms at 5 x background = 50 cps. With $K1 = 0.987 \text{ dK/K}$ ($p1 = -0.01317$), calculate $K2 = 0.9974 \text{ DKr/K}$ ($p2 = -0.00261$). Delta-P = 1056 pcm. $1056 / -10.75 = -98.2$ ppm change required. Therefore the time required for the 98.2 ppm dilution is $98.2 / 47.2 = 2$ hours 5 min. Difference in time based on use of Nomograph for RCS at normal pressure & temperature conditions. 'd' would only occur if count rate doubled in any 10 minute period. Assuming count rate increase is linear, for given dilution rate counts would change by 3 every 10 minutes.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Reactor Makeup Control system lesson plan		8	4,7,11
Source Range Nuclear instrumentation Lesson plan		6	6,10,11
Braidwood Curve Book Boron dilution rate nomograph			

Material Required for Examination Braidwood CURVE BOOK Figure 12.
 Question Source: New Question Modification Method:
 Question Source Comments:
 Comment Type Comment

The following conditions exist on Unit 1:

- Reactor power was 95% prior to the event
- A turbine runback resulted in rod insertion with control rods in AUTOMATIC
- Annunciator ROD BANK LO-2 INSERTION LIMIT (1-10-A6) is lit

The operators initiated an emergency boration per BWOA PRI-2 "Emergency Boration" and have verified control rods are now withdrawing. Why does the operator energize the PZR Backup Heaters?

This action...

- a. ensures PZR boron concentration equalization with RCS by increasing normal spray flow.
- b. counteracts RCS cooldown due the boration by the additional heat from the backup heaters.
- c. prevents loss of PZR level by increasing the volume of fluid maintained in the PZR.
- d. guarantees adequate subcooling margin is maintained by raising the saturation temperature of the PZR.

Answer a Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 004 K6.01 RO Value: 3.1 SRO Value: 3.3 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Title: Chemical and Volume Control System

KA Statement: Knowledge of the of the effect of a loss or malfunction on the following will have on the Chemical and Volume Control System:

Spray/heater combination in PZR to assure uniform boron concentration

Explanation of answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
BWOA Pri-2 Emergency Boration lesson plan		6	6
Reactor Makeup control system lesson plan		8	12

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist on Unit 1:

- A LOCA has occurred
- Actions of 1BwEP ES-1.3, 'Transfer To Cold Leg Recirculation, have been completed.
- During alignment, 1CV8804A, RH HX to CENT CHG Pumps Isolation Valve, failed to open and could NOT be manually opened.

What is the status of the ECCS system?

- a. The RHR discharge headers are cross-tied with only RHR Pump 1B running and supplying suction to the SI pumps and Centrifugal Charging pumps from the B train connection.
- b. The RHR discharge headers are cross-tied with both RHR pumps running and supplying suction to the SI pumps only from the B train connection. The Centrifugal Charging pumps are stopped.
- c. RHR Pump 1B is discharging through the B Train cold leg injection headers and supplying suction to the SI Pumps. RHR Pump 1A and the Centrifugal Charging pumps are stopped.
- d. RHR Pump 1B is discharging through the B Train cold leg injection headers and supplying suction to the SI pumps and Centrifugal Charging pumps. RHR Pump 1A is discharging through the A Train cold leg injection headers.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 005 K1.12 RO Value: 3.1 SRO Value: 3.4 Section: SYS RO Group: 3 SRO Group: 3

System/Evolution Residual Heat Removal System

KA Knowledge of the physical connections and/or cause-effect relationships between Residual Heat Removal System and the following: Safeguard pumps

Explanation of Answer CL recirc lineup has any ONE running RHR pump aligned to provide suction path to all other ECCS pumps (SI & CENT CHG). The discharge headers between RH trains are required to be separate so that the ONE running RH pump does not operate in runout condition.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Emergency Operating Procedures Loss of Reactor or secondary coolant/ BwEP 1, BwEP ES 1.1-1.4		11	10
Chp 58 Emergency Core Cooling system Lesson plan		10	5,7,8,14

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 25 Failure of Hx Outlet Valve

The following conditions exist on Unit 1:

- Unit is in MODE 4 during cooldown per 1BwGP 100-5 following unit shutdown 38 hours ago
- RCS temperature - 340°F
- RCS pressure - 345 psig
- PZR level - 33%
- RHR pump 1A is operating in Shutdown Cooling mode
- RH-618 A Hx Bypass Flow Control Valve is in MAN at 3000 gpm
- RH-606 A HX Flow Control Valve controller demand is at 20%
- CV-128 RHR Ltdn Flow Contr Valve demand is at 100%
- PCV-131 is in AUTOMATIC set to maintain 350 psig

A signal failure from the controller causes RH-606 to go fully closed. What is the system response to this failure without operator action?

- a. PCV-131 will throttle open due to lower RH discharge pressure.
- b. RCS pressure will increase due to RCS heatup.
- c. Pressurizer level will decrease due to increased letdown flow.
- d. RH-610 will throttle open due to lower RH flow.

Answer	d	Exam Level	R	Cognitive Level	Application	Facility:	Braidwood	ExamDate:	9/14/98
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KA:	005 K4.10	RO Value:	3.1	SRO Value:	3.1	Section:	SYS	RO Group:	3	SRO Group:	3
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System/Evolution Residual Heat Removal System

Knowledge of Residual Heat Removal System design feature(s) and or interlock(s) which provide for the following:
Control of RHR heat exchanger outlet flow

Explanation of Answer: RCS pressure will rise as fluid temperature increases due to loss of cooling flow through HX. IF flow decreases system pressure downstream may decrease this will cause PCV-131 to throttle close in an attempt to raise pressure

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
RHR Cooldown/ RH-1 Schematic	RH-1	1	
Chp 18Residual Heat Removal system		7	3,4,5,9

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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The following conditions exist on Unit 1:

- A plant heatup is underway
- MODE 3 has just been entered
- RCS pressure 450 psig

SI Accumulator 1C was drained below required level during the outage for repair work. System configuration has NOT allowed refilling the Accumulator until now. The SI Accumulator line is being flushed in accordance with BOP SI-14 "SI Accumulator Fill Line Flush" (Valve lineup includes: 1SI-8964, SI Test Lines to Radwaste Isolation Valve, and SI-8888, SI Pps to Accumulator Fill Valve, are open. 1SI 8821A, SI Pump to Cold Leg Isolation Valve, and 1SI 8802A, SI to Hot Leg 1A & 1D Isol valve are closed). SI pump 1A running. During the flushing, an inadvertent SI signal is generated.

What is the status of the ECCS based on the current alignment without operator action?

- a. 1B SI pump injection flow is directed to the RCS cold legs and 1A SI pump flow is directed to the Accumulator 1C fill line flush.
- b. 1A SI pump flow is directed to the 1C Accumulator fill line flush and 1B SI Pumps is in PULL-TO-LOCK.
- c. BOTH SI pump flows are directed to the RCS cold legs and to the Accumulator 1C fill line flush.
- d. BOTH SI pump flows are directed to the RCS cold legs ONLY.

Answer a Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 006 A2.13 RO Value: 3.9 SRO Value: 4.2 Section: SYS RO Group: 2 SRO Group: 2
 stem/Evolution Emergency Core Cooling System

KA Ability to (a) predict the impacts of the following on the Emergency Core Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
 Inadvertent SIS actuation

Explanation of Answer SI pumps are operable; SI8821A. remains closed; SI8888 and SI8964 remain open.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Plant Heatup BwGP 100-1	F.49 pg 30	11	
SI Accumulator Fill Line Flush BwOP SI-14		6	
Chp 58 Emergency Core Cooling system Lesson plan		10	6,9

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 27 10CFR50.46 Design Criteria

To meet the 10CFR50.46 criteria, the ECCS System is designed such that under accident conditions it will maintain...

- a. total hydrogen production from zirconium-water reaction below maximum value of 5%.
- b. maximum fuel temperature at the inside surface of the cladding NOT to exceed 2000°F.
- c. the core at least 5% dK/K shutdown to prevent an inadvertent return to criticality.
- d. fuel clad oxidation less than 17% of total clad thickness anywhere within the core.

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 006 K3.02 RO Value: 4.3 SRO Value: 4.4 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Emergency Core Cooling System

KA Knowledge of the effect that a loss or malfunction of the Emergency Core Cooling System will have on the following:
Fuel

Explanation of Answer Third selection addresses design criteria for reactivity control per CTS.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Chp 58 Emergency Core Cooling system Lesson plan	10CFR50/ 47	10	2

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type Comment

The following conditions exist on Unit 1:

- A LOCA has occurred
- 1B SI pump trips and cannot be restarted
- Transfer to Cold Leg recirculation is required
- RCS pressure is approximately 50 psig

What is the approximate total SI pump flow indicated on the main control board and how will this value change following transfer of BOTH trains of ECCS to cold leg recirculation?

Total Flow	Flow Change
a. 400 gpm	Decrease
b. 650 gpm	Increase
c. 800 gpm	Decrease
d. 1300 gpm	Increase

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 006 K6.03 RO Value: 3.6 SRO Value: 3.9 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Emergency Core Cooling System

KA Knowledge of the effect of a loss or malfunction on the following will have on the Emergency Core Cooling System:
 Safety Injection Pumps

Explanation of Answer SI pump design values provide for 650 gpm flow per pump @ 1300 psig and 1300 gpm @ 600 psig (or less).
 The flow from the pumps increases since the RH pumps are now providing a suction pressure of approximately 250 psig to the pumps instead of the lower pressure (30 psig or less) provided by the head associated with RWST level.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Chp 58, Emergency Core Cooling System Lesson plan		10	3, 8a

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 29 PRT conditions causing alarm/response

During shift turnover for Unit 1, the NSO notes the following parameters:

RCS Tave - 566.5°F
Pzr pressure - 2235 psig
Pzr level - 38.3%
PRT pressure - 4 psig
PRT level - 74%
PRT temperature - 98°F

One hour later when annunciator 1-12-A7, PRT LEVEL HIGH LOW alarmed, the NSO notes the following parameters:

RCS Tave - 566.2°F
Pzr pressure - 2233 psig
Pzr level - 38%
PRT pressure - 5.9 psig
PRT level - 81%
PRT temperature - 96°F

What condition resulted in the change in parameters?

- a. PRT PW Supply Inside Cnmt Isol Valve RY-8030 opened.
- b. PRT to GW Comp Isol Valve RY-469 failed closed.
- c. CVCS letdown relief valve CV-8117 lifted.
- d. PORV RY-455A opened and reclosed.

Answer a Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 2.4.50 RO Value: 3.3 SRO Value: 3.3 Section: SYS RO Group: 3 SRO Group: 3
 System/Evolution Pressurizer Relief Tank/Quench Tank System
 KA

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Explanation of Answer The only input provided that would give a level increase and a temperature decrease is the makeup from PW.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Pressurizer Relief Tank Filling and Venting BwOP RY-3		3	
PRT Level High low/ BwAR 1-12-A7		51E1	
Chp 14 Pressurizer lesson plan		9	13,14

Material Required for Examination

Question Source: New

Question Modification Method: Editorially Modified

Question Source Comments: Ginna 9/90 NRC Exam

Comment Type Comment

Question Topic Determination of effect of valve positioning

Unit 1 is operating at 100% power in MOL conditions. All systems are functioning normally with rod control in manual.

What is the effect on plant operations if instrument air supplied to the CVCS letdown Hx component cooling water outlet valve, TCV-CC-130 is lost?

TCV-CC-130 goes fully...

- a. shut and reactor power decreases due to boration in the CVCS demineralizers.
- b. shut and the CVCS demineralizers are automatically bypassed on temperature signal.
- c. open and reactor power increases due to deboration in the CVCS demineralizers.
- d. open and the CVCS demineralizers are automatically bypassed on temperature signal.

Answer C Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 008 A2.05 RO Value: 3.3 SRO Value: 3.5 Section: SYS RO Group: 3 SRO Group: 3

System/Evolution Component Cooling Water System

KA Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
Effect of loss of instrument and control air on the position of the CCW valves that are air operated

Explanation of Answer The CVCS letdown flow is overcooled and will give up boron to the resins in the CVCS demins (until a new equilibrium value of boron reached in demins).

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
of Instrument Air/ 1BwOA Sec-4	Table A Component clg	2	
Ch15a CVCS lesson plan Service Air/ Instrument Air Lesson plan	review quest 14	10 8	10,14 9

Material Required for Examination

Question Source: New

Question Modification Method:

Number(s) n

Question Source Comments:

Comment Type Comment

Question 3/ Spray using Normal and Aux Spray

What are the parameters and values used by the operator to ensure the temperature difference between the PZR and the spray fluid are within the specified limit(s) in the PRESSURE AND TEMPERATURE LIMIT REPORT when initiating PZR spray?

- a. For normal spray, the difference between RCS hot leg loop temperature and PZR vapor space temperature limit is 50°F, and for aux spray, the difference between Regenerative Hx charging inlet temperature and PZR vapor space limit is 320°F.
- b. For normal spray, the difference between RCS cold leg loop temperature and PZR vapor space temperature limit is 50°F, and for aux spray, the difference between Regenerative Hx charging outlet temperature and PZR vapor space limit is 320°F.
- c. For normal spray, the difference between RCS hot leg loop temperature and PZR vapor space temperature limit is 320°F, and for aux spray, the difference between Regenerative Hx charging inlet temperature and PZR vapor space limit is 320°F.
- d. For normal spray, the difference between RCS cold leg loop temperature and PZR vapor space temperature limit is 320°F, and for aux spray, the difference between Regenerative Hx charging outlet temperature and PZR vapor space limit is 320°F.

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 010 A1.08 RO Value: 3.2 SRO Value: 3.3 Section: SYS RO Group: 2 SRO Group: 2
 System/Evolution Pressurizer Pressure Control System
 KA Ability to predict and/or monitor changes in parameters associated with operating the Pressurizer Pressure Control System controls including:
 Spray nozzle DT

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Pressurizer Temperature Limit Surv/ 1BwOS 4.9.2-1			
Pressurizer Spray Water Temperature Differential Limit surv/ 1BwOS 4.9.2-2			
1BwGP 100-1 Plant heat up lesson plan		12	1,2,3
Chp 14 Pressurizer lesson plan		9	7,8

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: Kewaunee 2/94 NRC Exam

Comment Type Comment

Question 32 Evaluation of Pzr conditions

The following conditions exist on Unit 1:

- A load reject from 100% power has occurred
- Reactor power - 80%
- Pzr level - 56%
- Pzr vapor temperature - 655°F
- Pzr liquid temperature - 653°F
- RCS Tave - 578°F

What is the current status of the Pressurizer based on given conditions?

- a. Backup and proportional heaters are fully on.
- b. Proportional heaters are modulated on.
- c. Pzr spray valves have modulated open.
- d. Pzr spray valves and Pzr PORVs are open.

Answer c Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 010 K5.01 RO Value: 3.5 SRO Value: 4.0 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Pressurizer Pressure Control System

KA Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Pressure Control System:
Determination of condition of fluid in PZR, using steam tables

Explanation of Answer At 655°F, saturation pressure is 2272 psig. At this pressure, with current PZR level deviation <5% of program level(53%), the sprays are the only component "on".

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
P Pressure Control/ RY-2	Pzr Pressure Setpoints		
Chp 14 Pressurizer lesson plan		9	5,6,7
Steam tables	Saturation table		

Material Required for Examination Steam Tables

Question Source: Facility Exam Bank Question Modification Method: Concept Used

Question Source Comments: Braidwood 1997 NRC exam

Comment Type Comment

Question **33** Pzr Level Reactor Trip

The following conditions exist on Unit 1 with all controls in normal lineup:

- Reactor power - 30% stable
- RCS Tave - 564.5°F
- Pzr pressure - 2230 psig
- Pzr level - 36% (LI-459), 37% (LI-460), 36% (LI-461)
- Pzr LVL CONT CH SELECT is in 459/460 position

The pressurizer level controller 1LK-459 output fails low. What automatic actions will occur as a result of this failure assuming NO operator action taken?

- a. Pzr level will NOT change due to LT-460 being the controlling channel
- b. The reactor will trip on high Pzr level due to letdown isolation.
- c. Pzr level will control at 25% due to low output from the controller.
- d. Pzr level will control at 60% due to low output from the controller.

Answer **b** Exam Level **B** Cognitive Level **Comprehension** Facility: **Braidwood** ExamDate: **9/14/98**

KA: 011 K1.04 RO Value: 3.8 SRO Value: 3.9 Section: **SYS** RO Group: **2** SRO Group: **2**

System/Evolution **Pressurizer Level Control System**

KA Knowledge of the physical connections and/or cause-effect relationships between Pressurizer Level Control System and the RPS

Explanation of Answer NOTE that this failure is like the failure of the controlling level channel high in that charging flow falls to minimum. At 17% level, letdown isolates charging continues at minimum (52 gpm) and Pzr level rises to high level trip setpoint.).

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Level Control schematic	RY-3 Pzr level setpts	2	
Chp14 Pressurizer lesson plan		9	21

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments:

Comment Type Comment

Question **34** Operation of BOTH Bypass Trip Breakers

The following conditions exist on Unit 1:

- Mode 3 NOT NOP with reactor trip breakers (RTA and RTB) closed
- Testing of reactor trip bypass breakers underway
- Reactor bypass breaker B (BYB) is racked in and closed
- An operator begins to perform test with reactor bypass breaker A (BYA).

What occurs as the operator operates the breaker BYA?

When reactor bypass breaker BYA is...

- locally closed, breaker BYB will trip. RTA and RTB remain closed.
- racked in to the CONNECT position, breaker BYB will trip. RTA and RTB remain closed.
- locally closed, all reactor trip and bypass breakers will trip.
- is racked in to the CONNECT position, all reactor trip and bypass breakers will trip.

Answer C Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 012 A3.07 RO Value: 4.0 SRO Value: 4.0 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Reactor Protection System

KA Ability to monitor automatic operations of the Reactor Protection System including:
Trip breakers

Explanation of Answer Closure of the second BYB results in SPSS generating a GENERAL WARNING on both trains which would open all trip and bypass breakers.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
setpoints Schematic	EF-2 Rx Trip Byp brkr trips	5	
Ch 60a SSPS lesson plan		3	6,9

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments:

Comment Type Comment

Question 35 Input that can be bypass & condition

The following conditions exist on Unit 2:

- Unit shutdown is in progress
- Reactor power - 20%
- RCS Tave - 562°F
- Pzr pressure - 2235 psig
- Pzr level - 32%
- First stage turbine pressure channel PT-506 fails high

What affect does this failure have on operations as unit shutdown is continued, if NO action is taken for the channel failure?

- At 10% power, the reactor will trip if the SR MAN BLOCK switches are taken to RESET.
- At 9% power, the reactor will trip if an RCP trips.
- At 7% power, the reactor will trip if the TURBINE TRIP pushbuttons are depressed.
- At 5% power, the reactor will be manually tripped as required during a normal shutdown.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 012 A4.03 RO Value: 3.6 SRO Value: 3.6 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Reactor Protection System

KA Ability to manually operate and/or monitor in the control room:
Channel blocks and bypasses

Explanation of Answer PT-506 failure results in P13 interlock NOT clearing when turbine power falls below 10%. This also feeds into P7 "AT POWER TRIPS" interlock also remains active. Trips affected: 1) 2 loop loss of flow, 2) Pzr low press, 3) Pzr high level, 4) RCP brkr open, 5) RCP UV, 6) RCP UF. At 10% power, the SR NIS should still be auto blocked by P-10 (active). The turbine is normally tripped from ~65 Mwe at 5% power per BwGP.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Power Descension /1BwGP 100-4	note step F.27	16	
ESF Setpoints/Schematic	EF-1/ Permissive Rx Trip	4	
Ch60b/ Reactor Protection system		6	4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist on Unit 1:

- Power range NIS reading - 100%
- Tcold - 553°F
- That - 608°F
- RCS total flow - 372,000 gpm
- Pzr pressure - 2245 psig
- Pzr level - 69%

How does the setpoint for Over Temperature Delta-T (OTdT) change when a listed parameter is changed? (Consider each change individually)

The setpoint...

- a. increases if Power range NIS output rises to 102%.
- b. decreases if total reactor flow increases to 370,000 gpm.
- c. increases if pressurizer pressure decreases to 2235 psig.
- d. decreases if the That rises to 612°F.

Answer d Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 012 K5.01 RO Value: 3.3 SRO Value: 3.8 Section: SYS RO Group: 2 SRO Group: 2

Title: Reactor Protection System

System/Evolution Statement:

KA Knowledge of the operational implications of the following concepts as they apply to the Reactor Protection System:
DNB

Explanation of a - NIS input is only for exceeding +/- delta-I; b - Flow affects when DNB occurs, but is NOT an input to OTdT;
Answer c - Pressurize rise increases OTdT. That input to dT power for OTdT determination Number(s) n

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
ESF Setpoints/ EF-2	OTDT	5	
CH 60b/ RPS lesson plan		6	3,4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question **37** CNMT Spray/Phase B

A heatup is in progress on Unit 1.

At 0700, the following conditions are noted:

- RCS pressure - 1750 psig
- RCS temperature - 480°F
- S/G pressures - 565 psig

At 0730, the following conditions are noted:

- RCS pressure - 1850 psig
- RCS temperature - 485°F
- S/G pressures - 593 psig

If the current trend continues, the FIRST event that the operators should expect to see is the...

- a. Pzr PORVs open
- b. MSIVs close
- c. Pzr sprays open.
- d. S/G PORVs open

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

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

013 Engineered Safety Features Actuation System

Knowledge of Engineered Safety Features Actuation System design feature(s) and or interlock(s) which provide for the following:

K4.03 Main Steam Isolation System 3.9 4.4

Explanation of RCS (Pzr) pressure rises above the P-11 setpoint (1930 psig), which provides permissive for SI/Main Steam

Answer Line Isolation on low S/G pressure, and S/G pressure is less than MSL isolations.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
ESF Setpoints/ EF-2	CS/Phase B sig	5	
CS/ MCB indications/ CS-1, CS-2	CS Actuation sig	3	
Chp 61 ESF lesson plan		5	7,8

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist on Unit 2:

- RCS temperature - 340°F
- RCS pressure - 900 psig
- All MSIVs for the S/Gs are closed
- The MSIV Bypass valves are open
- The FW-035s, Feedwater Tempering Isolation Valves, are open
- The FW-034s, Feedwater Tempering Flow Control Valves, are closed (opened periodically for level control)
- Feedwater pump 2C is reset and latched on turning gear
- The Start Up Feedwater pump is running

The level in the S/G 2B rises to 90%. How is the plant affected?

- a. No actuation occurs because of the position of the MSIVs.
- b. The 2C Feedwater pump and Start Up Feedwater pump trip.
- c. The 2C Feedwater pump trips and FW-035 valves close.
- d. The 2C Feedwater pump and Start Up Feedwater pump trip, the FW-035 valves close, and the MSIV Bypass valves close.

Answer C Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 013 K4.13 RO Value: 3.7 SRO Value: 3.9 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Engineered Safety Features Actuation System

KA Knowledge of Engineered Safety Features Actuation System design feature(s) and or interlock(s) which provide for the following: MFW isolation/reset

Explanation of Answer Having Loop Isolation Stops closed does not defeat P-14.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Feedwater simple/ FW-1	FWI signals	4	
SGWLC/ FW-2	S/U Flowpaths	0	
Chp61 ESF lesson plan		5	7

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 39 ROD BOTTOM Alarm operation

During a reactor startup, when does the ROD AT BOTTOM alarm become active for each control bank?

The alarm will actuate for a dropped rod for...

- a. any Control Bank whenever Control Bank A DRPI output is above 9 steps.
- b. each Control Bank whenever that Control Bank demand position is above 3 steps.
- c. each Control Bank whenever that Control Bank DRPI output is above 9 steps.
- d. Control Banks A, B and C whenever their Control Bank demand position is above 9 steps, and for Control Bank D whenever Control Bank D demand position is above 3 steps.

Answer C Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 2.4.31 RO Value: 3.3 SRO Value: 3.4 Section: SYS RO Group: 2 SRO Group: 1

System/Evolution Rod Position Indication System

KA Knowledge of annunciators alarms and indications, and use of the response instructions.

Explanation of Answer Note that the ROD BOTTOM comes directly from the DRPI unit with a setpoint of 9 steps; the alarm actuates when rod position is detected at 3 steps (or less).

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
ROD at Bottom/ 1BWAR 1-10-E6		2	
Chp 29 rod Position Indication sys Lesson plan		9	4,5

Material Required for Examination

Question Source: New Question Modification Method: Significantly Modified

Question Source Comments: Millstone 3 11/90 NRC Exam

Comment Type Comment

Question **40** SR NIS discriminator failure

How would the failure of the pulse height discriminator to a low value affect the indication of the affected Source Range channel?

The output would increase due to...

- a. electronic filtering which narrows the pulse height window.
- b. failure in removing the higher amplitude neutron generated pulses.
- c. increased gamma interaction inside the detector.
- d. counting of the gamma generated pulses and decay-alpha generated pulses.

Answer d Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 015 A2.02 RO Value: 3.1 SRO Value: 3.5 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Nuclear Instrumentation System

KA Ability to (a) predict the impacts of the following on the Nuclear Instrumentation System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Faulty or erratic operation of detectors or compensating components

Explanation of Answer Pulse height discriminator used to set window to detect those pulses with energy level high enough to be from event associated with neutron detection. Gamma and other interactions such as the alpha decay of fission product daughters is of lower height (energy) and discriminator normally electronically removes.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Source Range Detector schematic	NI-4	4	
Chp 31 Source Range Nuclear Inst		6	3

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 4/ SR NIS - loss of control power

The following conditions exist on Unit 1:

- RCS at NOT NOP
- Reactor trip breakers - closed
- Source Range readings:
 - N31 - 18 cps
 - N32 - 22 cps

What indication would the operator observe if Control Power was lost to the N31 Drawer?

The N31 meter would read...

- a. downscale, the associated drawer bistable lamps NOT lit, and reactor trip breakers closed.
- b. downscale, the associated drawer bistable lamps lit, and reactor trip breakers open.
- c. 18 cps, the associated drawer bistable lamps NOT lit, and reactor trip breakers closed.
- d. 18 cps, the associated drawer bistable lamps lit, and reactor trip breakers open.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 015 K2.01 RO Value: 3.3 SRO Value: 3.7 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Nuclear Instrumentation System

KA Knowledge of electrical power supplies to the following:
NIS channels, components, and interconnections

Explanation of answer Control power loss affects bistables which trip but NOT drawer instrument indication which is from Instrument Power source.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Source Range Detector Schematic	NI-4 loss of Control power	4	
Ch 31 Source Range Nuclear Inst		6	8.b

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 42 Eval for 1/M - Eightfold increase

The following conditions exist on Unit 1:

- A reactor startup is about to be performed
- All shutdown banks are fully withdrawn
- All control banks are fully inserted
- An ECC records the following:
 - Predicted Critical Position (ECP) - 130 steps on CBD
 - Max rod position - 231 steps on CBD
 - Min rod position - 58 steps on CBD

The following parameters were recorded during the rod withdrawal:

ROD HEIGHT	N31 cps	N32 cps
0 on CBA	25	23
178 on CBA	34	31
178 on CBB	80	82
178 on CBC	200	162
80 on CBD	237	184
92 on CBD	260	245

When was the first time the operator was required to determine the Predicted Critical Position?

- a. At 50 steps on CBA, with N32 as the designated Source Range detector.
- b. At 47 steps on CBC, with N31 as the designated Source Range Detector.
- c. At 178 steps on CBC, with N31 as the designated Source Range detector.
- d. At 80 steps on CBD, with N32 as the designated Source Range detector.

Answer c Exam Level R Cognitive Level Comprehension Facility: Byron ExamDate: 9/14/98
 Tier: Plant Systems RO Group: 1 SRO Group: 1

015 Nuclear Instrumentation System

K5. Knowledge of the operational implications of the following concepts as they apply to the Nuclear Instrumentation System:

K5.06 Subcritical multiplications and NIS indications 3.4 3.7

Explanation of During reactor SU, hold point for ICRR determination is performed for each Control Bank at 50 steps withdrawn. The actual determination of Predicted Critical Position is required at the eight-fold count increase on highest reading SR. Holdpoint occurs on CBC @178. \$& steps on CBC is the 0% power RIL value.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
1BwGP 100-2 Reactor Startup	1BwGP 100-2A1	12	
1BwGP 100-2A1 Lesson plan		-13	2

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

The following conditions exist on Unit 1:

- Reactor power - 50%
- RCS Tave - 570°F (A); 569°F (B); 569°F (C); 570°F (D)
- RCS Thot - 585°F (A); 584°F (B); 583°F (C); 585°F (D)
- RCS Tcold - 555°F (A) 554°F (B); 555°F (C); 555°F (D)
- Pzr pressure - 2235 psig
- Pzr level - 43 %

If loop B Thot output channel fails LOW, what is the response of Pzr level ?

Pressurizer level will...

- a. increases to 60%.
- b. remains the same.
- c. decreases to 25%.
- d. decreases to the letdown isolation setpoint.

Answer **b** Exam Level **B** Cognitive Level **Comprehension** Facility: **Braidwood** ExamDate: **9/14/98**
 KA: 016 K3.02 RO Value: 3.4 SRO Value: 3.5 Section: **SYS** RO Group: **2** SRO Group: **2**
 System/Evolution **Non-Nuclear Instrumentation System**
 KA Knowledge of the effect that a loss or malfunction of the Non-Nuclear Instrumentation System will have on the following:
 PZR LCS
 Explanation of answer Tnot fails to 510°F. With loop Tcold of 537°F, loop Tave is now 524°F. Auctioneered HIGH Tave is used for Pzr level program.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
PZR Level Control Schematic	RY-3	2	
1BwOA Inst-2 lesson plan		15	1
chp 12 RCS lesson plan		8	13

Material Required for Examination

Question Source: Facility Exam Bank Question Modification Method: Concept Used

Question Source Comments: Zion 2/92 NRC Exam (along with several others). Change includes failure of Thot loop, failure low and conditions instead of dual condition.

Comment Type Comment

Question **44** CETC failure effect on Subcooling Monitor/Iconic Display

With Unit 1 at 100% power and with normal operating parameters, how would the failure of the HOTTEST Core Exit Thermocouple affect the reading of subcooling margin on the SPDS Iconics (CETC/SMM display) for each of the two situations below:

- Situation 1 - The CETC output fails high slowly
- Situation 2 - The CETC output fails low slowly

- a. Situation 1: Subcooling margin will decrease to saturation then indicate superheated, and return to normal when CETC output reaches 2300°F.
Situation 2: Subcooling margin will increase, then stabilizes when the CETC output is smaller than TEN other TCs.
- b. Situation 1: Subcooling margin will decrease to saturation then indicate superheated, and return to normal when CETC output reaches 1200°F.
Situation 2: Subcooling margin will remain constant.
- c. Situation 1: Subcooling margin will increase to saturation then indicate superheated, and return to normal when CETC output reaches 1200°F.
Situation 2: Subcooling margin will decrease, then stabilizes when the CETC output is smaller than TEN other TCs.
- d. Situation 1: Subcooling margin will Increase to saturation then indicate superheated, and return to normal when TC output reaches 2300°F.
Situation 2: Subcooling margin will remain constant.

Answer a Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 017 K4.01 RO Value: 3.4 SRO Value: 3.7 Section: SYS RC Group: 1 SRO Group: 1

Item/Evolution In-Core Temperature Monitor System

KA Knowledge of In-Core Temperature Monitor System design feature(s) and or interlock(s) which provide for the following: Input to subcooling monitors

Explanation of Answer Fail high - Since it is initially the highest, its input will remain active in average until high setpoint reached at 2300°F. Fail low - subcooling margin will slightly increase as temperature falls and input to average remains valid. When it reaches the 11th highest value, the subcooling margin will stabilize and remain constant (assuming other 10 inputs do not change).

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Chapter 34b Inadequate Core Cooling Detection		7	5,6

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: Braidwood 1997 NRC Exam. Difference in all answer choices - similar premise in theory, but different wording.

Comment Type Comment

Question ⁴⁵ RCFC operations requirements

The following conditions exist on Unit 2:

- RCS Temperature - 342°F
- Pzr pressure - 375 psig
- 2A, 2B, and 2D RCFCs are operating in high speed
- Unit 2 RCFC Dry Bulb temperatures are recorded as follows:
 - 2A RCFC - 119°F
 - 2B RCFC - 118°F
 - 2C RCFC - 127°F
 - 2D RCFC - 121°F

Which of the following identifies the equipment status and actions for the above conditions?

- a. RCFC 2C must be started because the average of ALL the RCFC temperatures exceeds the limit.
- b. RCFC 2C must be started because ONE of the operating RCFCs temperatures is above the limit.
- c. NO action is necessary because ALL temperatures are within the required limit.
- d. NO action is necessary because the average temperature of ALL operating RCFCs is below the limit.

Answer d Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 2.1.32 RO Value: 3.4 SRO Value: 3.8 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Containment Cooling System

KA

Ability to explain and apply all system limits and precautions.

Explanation of answer Limits on CNMT temperature determined by average of temperatures for OPERATING RCFC outlet temps.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
RCFC Start up 1BwOP VP-5			
U2 Mode 1,2,3 shiftly daily Op surv			
2BwOS-0.1-1,2,3			
chp 42 Containment Vent system lesson plan		4	6, 10a

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Topic

Question **46** Sequence for securing CNMT Spray

The following conditions exist on Unit 1:

- A LOCA has occurred
- Transition has been made to BwEP ES-1.3 "Transfer To Cold Leg Recirculation"
- Containment Spray actuated due to high containment pressure
- All systems and components operating as expected

What conditions allow for termination of Containment Spray?

- a. ONE pump is stopped when containment pressure is less than 15 psig. The other pump is stopped when RWST LO-3 level is reached.
- b. ONE pump is stopped when containment pressure is less than 20 psig. The other pump is stopped after it has operated for a period of at least TWO hours
- c. BOTH pumps are stopped when containment pressure is less than 15 psig and have operated for a period of at least TWO hours.
- d. BOTH pumps are stopped when containment pressure is less than 20 psig and RWST LO-3 level is reached.

Answer C Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 026 A2.08 RO Value: 3.2 SRO Value: 3.7 Section: SYS RO Group: 2 SRO Group: 1

System/Evolution Containment Spray System

KA Ability to (a) predict the impacts of the following on the Containment Spray System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
Safe securing of containment spray when it can be done)

Explanation of Answer
Title:

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Containment Spray Schematic	CS-1/ CS term	3	
Loss of Reactor or Sec Coolant/1BwEP-1		1B WOG-1B	
Ch 59 Containment Spray sys Lesson plan		6	14

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 47 Pump operation interlocks

The following conditions exist on Unit 1:

- LOCA is in progress
- Containment pressure - 17 psig
- Containment Spray actuated due to high containment pressure
- Containment Spray signal has been reset
- The actions of BwEP ES-1.3 "Transfer To Cold Leg Recirculation" have been completed
- Offsite power is then lost and the D/G output breakers have just closed onto ESF buses

How are the Containment Spray Pumps re-started?

- a. The pumps will auto start 15 seconds following closure of the D/G output breakers.
- b. The pumps will auto start 40 seconds following closure of the D/G output breakers.
- c. If the operator immediately places the CS & PHASE B ISOL switches for both trains to ACTUATE, the pumps will auto start 15 seconds following closure of the D/G output breakers.
- d. If the operator immediately places the PP 1_ TEST switches for both pumps in TEST, the pumps will auto start 40 seconds following closure of the D/G output breakers.

Answer C Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 026 A4.01 RO Value: 4.5 SRO Value: 4.3 Section: SYS RO Group: 2 SRO Group: 1

System/Evolution Containment Spray System

KA Ability to manually operate and/or monitor in the control room:
CSS controls

Explanation of Answer If the AUTO actuation input signal is absent and actuation input has been reset, manual actuation is required to get equipment restarted following a LOSEP.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Chp 59 Containment spray sys lesson plan		6	8,9

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 48 Charcoal Filters response to deluge

Annunciator 0-33-C3, FILTER 1VP05FA TEMPERATURE HIGH, alarms in the Control Room while 1VP02CA CNMT Charcoal Filter Fan is operating. The alarm condition is verified locally.

Which of the following describes the actions taken and/or the system response for the Containment ventilation System?

- a. The deluge valve FP244A will automatically open and the fan will automatically stop.
- b. The control room operator will open the deluge valve FP244A and the local operator will then stop the fan.
- c. The local operator will open the deluge valve FP244A and the fan will automatically stop.
- d. The local operator will open the deluge valve FP244A and the control room operator will then stop the fan.

Answer C Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 027 A4.03 RO Value: 3.3 SRO Value: 3.2 Section: SYS RO Group: 3 SRO Group: 2

System/Evolution Containment Iodine Removal System

KA Ability to manually operate and/or monitor in the control room: CIRS fans

Explanation of Answer Operation of fp components associated with charcoal filter is local. But fan trips when deluge system activated.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Filter 1VP05FA Temperature High /1BwaR 1VP01j-1-A1		1	
chp 42 Containment vent 7 purge		4	8

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist:

- Unit 1 - 20% power with load increase in progress
- Unit 2 - MODE 5 following refueling outage
- Unit 2 Spent Fuel Pool Cooling Loop is in service.
- Spent Fuel Pool Pump 1FC01P is OOS.

Which of the following is allowed under this situation?

Alignment and operation of...

- a. both Unit 1 RWST purification and Unit 2 RWST purification with flow through the Unit 2 Spent Fuel Pool Demineralizer and Unit 2 Spent Fuel Pool Filter.
- b. Unit 1 Spent Fuel Pool purification and Unit 1 RWST purification with flow through the Unit 1 Spent Fuel Pool Demineralizer and Unit 1 Spent Fuel Pool Filter.
- c. Unit 2 RWST purification with flow through the Unit 1 Spent Fuel Pool Filter and return to Unit 2 RWST.
- d. Unit 2 RWST purification with flow through the Unit 2 Spent Fuel Pool Demineralizer and Unit 2 Spent Fuel Pool Filter.

Answer d Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 033 K1.05 RO Value: 2.7 SRO Value: 2.8 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Spent Fuel Pool Cooling System

KA Knowledge of the physical connections and/or cause-effect relationships between Spent Fuel Pool Cooling System and the RWST

Explanation of Answer The lineup allows Unit 2 only to be used for Unit 2 RWST cleanup. Only one unit RWST can be aligned at time due to common input path via Refueling Water Purification Pumps. With the cooling loop inservice only, the Unit's RWST may be aligned through the same Unit's, demin and filter train. Simultaneous use of Demin/filter for the same Unit's SFP and RWST is NOT allowed due to concerns of draining RWST.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
S/U purification sys to purify or Recirculate the RWST/ BwOP FC-7		7	
Fuel Pool Cooling Schematic	FC-1	3	
Chp 51 Spent Fuel Pool Cooling And Cleanup		5	3

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 50 Steam Dump input malfunction

The following conditions exist on Unit 1:

- Reactor power was 65% when the turbine tripped
- An ATWS occurred
- The reactor tripped 15 seconds later when B reactor trip breaker was locally opened
- Reactor trip breaker A is failed closed
- RCS Tave - 559°F
- Pzr pressure - 2255 psig
- Steamline header pressure - 1100 psig
- No controls other than control rods and boration controls have been operated

What is the status of the Steam Dump valves?

Steam Dumps are...

- a. modulated open due to steam header pressure.
- b. modulated open due to Tave above no-load Tave.
- c. closed because Tave is NOT greater than 3°F above Tref.
- d. closed because the dumps are NOT armed.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 041 A3.02 RO Value: 3.3 SRC Value: 3.4 Section: SYS RO Group: 3 SRO Group: 3

System/Evolution Steam Dump System and Turbine Bypass Control

KA Ability to monitor automatic operations of the Steam Dump System and Turbine Bypass Control including: RCS pressure, RCS temperature, and reactor power

Explanation of Answer The "A" reactor trip breaker provides the arming signal for dumps on normal reactor trip. Since "A" RTB is still closed, the steam dumps respond to event like load rejection, with C-7 load rejection (10% load decrease in 2 minutes sensed on PT-506) arming the dumps. Since the "B" RTP was opened, the steam dump controller does operate on the plant trip controller (No load Tave compared to Auct Hi Tave).

Reference Title/Facility Reference Number	Section/Page	Revisin	L. O.
Steam Dumps/ Schematic	MS-4	4	
Chp 24 Steam Dumps Lesson Plan		7	3,4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 51 Turbine Control response to Failed Impulse Channel

The following conditions exist on Unit 1:

- Reactor power 28%
- All systems normal
- Turbine EHC Panel settings:
 - Turbine REFERENCE DEMAND - 580 MW
 - Turbine REFERENCE - 330 MW
- The GO pushbutton is LIT

What would be the DEHC System response to a slow failure to ZERO for the turbine impulse pressure channel that feeds into the DEHC?

Turbine load will...

- a. decrease until the difference between REFERENCE and impulse pressure exceeds 30%, the operator would then be alerted to select MANUAL control.
- b. decrease until the difference between REFERENCE DEMAND and impulse pressure exceeds 30%, then load will stabilize in MANUAL control.
- c. increase until the difference between REFERENCE and impulse pressure exceeds 30%, then load will stabilize in MANUAL control.
- d. increase until the difference between REFERENCE DEMAND and impulse pressure exceeds 30%, the operator would then be alerted to select MANUAL control.

Answer C Exam Level R Cognitive Level Comprehension Facility: B...idwood ExamDate: 9/14/98
 : 045 K1.20 RO Value: 3.4 SRO Value: 3.6 Section: SYS RO Group: 3 SRO Group: 3

System/Evolution Main Turbine Generator System

KA Knowledge of the physical connections and/or cause-effect relationships between Main Turbine Generator System and the Protection system

Explanation of Answer When the difference between actual load and turbine impulse pressure (IMP IN) channel exceeds, circuit AUTO transfer impulse feedback to IMP OUT

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
TV/GV Control/ schematic	EHC-3/ Impulse	1	
Chp 37a Main turbine Control And Protection		5	52

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Question **52** S/G Level program - low power

The following conditions exist on Unit 1:

- Reactor power 35%
- All systems normal

What failure would cause an INITIAL decrease in feedwater flow to all S/Gs?

- a. Turbine first stage impulse pressure PT-505 fails low.
- b. Main steamline pressure PT-507 fails low.
- c. Turbine first stage impulse pressure PT-506 fails low.
- d. Main feedwater header pressure PT-508 fails low.

Answer **b** Exam Level **B** Cognitive Level **Comprehension** Facility: **Braidwood** ExamDate: **9/14/98**
 KA: **2.1.7** RO Value: **3.7** SRO Value: **4.4** Section: **SYS** RO Group: **1** SRO Group: **1**
 System/Evolution **Main Feedwater System**

KA
 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Explanation of Answer **PT-507 fails low causes feed pump speed to decrease which reduces FW pressure. This would initially result in a decrease of flow to all S/Gs.**

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Fw EH controls/schematic	EHC-6/ DP	1	
Chp 27 SGWLC		6	16

Material Required for Examination
 Question Source: **New**

Question Modification Method:

Question Source Comments:
 Comment Type **Comment**

Question 53 Effect of failure of S/G steam pressure channel

The following conditions exist on Unit 1:

- Reactor power 100%
- All systems normal
- FT-512 selected for steam flow input into SGWLC for S/G 1A

What is the effect of the pressure transmitter associated with FT-512 failing low?

1A S/G level will decrease, ..

- a. feed pump speed will decrease and S/G level will decrease below the LO-2 setpoint.
- b. feed pump speed is unaffected, and S/G level will return to normal.
- c. feed pump speed will increase and S/G level will return to normal.
- d. feed pump speed is unaffected, and S/G level will decrease below LO-2 setpoint.

Answer a Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 059 K1.04 RO Value: 3.4 SRO Value: 3.4 Section: SYS RO Group: 1 ;RO Group: 1

System/Evolution Main Feedwater System

KA Knowledge of the physical connections and/or cause-effect relationships between Main Feedwater System and the following: S/GS water level control system

Explanation of Answer Steam flow is output to summator for FW control system program Delta-P. Delta-P program will decrease causing feed pump speed and FW header pressure to decrease.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
FW EH controls/ schematic	EHC-6/DP	1	
SGWLC schematic	FW-2/ 512 loop	0	
27 SGWLC lesson plan		6,	16

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist on Unit 1:

- The reactor tripped from an at-power condition
- An undervoltage condition exists on RCP 1C bus
- Power Range NIS channel N42 failed at 100% on the trip
- ESF bus 141 undervoltage occurred
- 1A D/G automatically started and ACB 1413 is closed
- S/G levels lowest readings were - 19% (A); 25% (B); 22% (C); 20% (D)

What is the status of the Auxiliary Feedwater (AF) Pumps on Unit 1 for these conditions at ONE minute following the trip?

- a. Both AF pumps are running.
- b. The 1A AF pump is running and the 1B AF pump is NOT running.
- c. The 1B AF pump is running and the 1A AF pump is NOT running.
- d. NO AF start signal is initiated.

Answer **b** Exam Level **B** Cognitive Level **Comprehension** Facility: **Braidwood** ExamDate: **9/14/98**

KA: 061 A3.01 RO Value: 4.1 SRO Value: 4.2 Section: **SYS** RO Group: **1** SRO Group: **1**

System/Evolution **Auxiliary / Emergency Feedwater System**

KA Ability to monitor automatic operations of the Auxiliary / Emergency Feedwater System including: AFW startup and flows

Explanation of Answer SG levels are above AF actuation setpoints and the motor driven AF pump starts on the detected undervoltage.

Source Title/Facility Reference Number	Section/Page	Revision	L. O.
Aux Feedwater System		2	5
Chp 26 AFW sys lesson plan		9	3,5
Chp 9 EDG lesson plan		7	7

Material Required for Examination

Question Source: **New**

Question Modification Method:

Question Source Comments:

Comment Type **Comment**

Question 55 AFW flow requirements for cooldown

In accordance with the BEPs, which of the following describes the MINIMUM AFW pump flow and S/G configuration necessary to remove all of the reactor decay heat load following a reactor trip from 102% power to preclude entry into loss of heat sink RED path entry?

- a. The 1A AF pump supplying 480 gpm to at least ONE S/G with S/G blowdown manually isolated.
- b. The 1B AF pump supplying 245 gpm to each of TWO S/G with S/G blowdown in service.
- c. The 1A AF pump supplying 170 gpm flow to each of THREE S/Gs with S/G blowdown manually isolated.
- d. The 1B AF pump supplying 130 gpm flow to each of FOUR S/Gs with S/G blowdown in service.

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 061 K5.02 RO Value: 3.2 SRO Value: 3.6 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Auxiliary / Emergency Feedwater System

KA Knowledge of the operational implications of the following concepts as they apply to the Auxiliary / Emergency Feedwater System: Decay heat sources and magnitude

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
AFW system lessson plan ch26		9	1,11

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: Comanche Peak 11/93 NRC Exam

Comment Type Comment

The following conditions exist on Unit 1:

- Reactor power - 100%

investigation has located a ground on the 125 VDC Normal supply to the 1A D/G. What action is required to transfer DC Control Power to the reserve source?

The Reserve power breaker from...

- DC 111 will be closed after opening the Normal power breaker and the Reserve power breaker at the D/G control panel.
- DC 111 will be closed after swapping the no-blow link at the Normal and Reserve power fuse blocks at the D/G control panel.
- DC 112 will be closed after opening the Normal power breaker and the Reserve power breaker at the D/G control panel.
- DC 112 will be closed after swapping the no-blow link at the Normal and Reserve power fuse blocks at the D/G control panel.

Answer b Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 2.1.30 RO Value: 3.9 SRO Value: 3.4 Section: SYS RO Group: 2 SRO Group: 1
 System/Evolution D.C. Electrical Distribution

KA Ability to locate and operate components, including local controls.

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
125 VDC system/schematic	DC-1	0	
DC Control power transfer from Normal to reserve source/ BwOP-DC-6A1		51	
Chp 8a 125 VDC lesson plan		6	4,6

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 57 Sequencing of ESF pumps - SI & SI w LOP

Unit 1 was being synchronized to the grid when the following occurred:

- Trip of 345 KV breakers resulted in deenergizing the SATs
- A steamline break occurred that resulted in containment pressure reaching 20 psig 20 seconds after the D/Gs output breakers have closed

When would the 1A SX pump re-start?

- a. Following start of the 1A CS Pump.
- b. Between the start of the 1A CV pump and the 1A RH pump.
- c. Between the start of 1A CC pump and the 1A AF pump.
- d. Coincident with the starting of the 1A and 1C RCFCs.

Answer C	Exam Level B	Cognitive Level Memory	Facility: Braidwood	ExamDate:	9/14/98
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KA: 064 A3.07	RO Value: 3.6	SRO Value: 3.7	Section: SYS	RO Group: 2	SRO Group: 2
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System/Evolution Emergency Diesel Generators

KA Ability to monitor automatic operations of the Emergency Diesel Generators including: Load sequencing

Explanation of Answer The SX pump would be started in this case by the SI signal which overrides the UV condition. The SX pump starts in following sequence: CV (0 sec); SI ((5 sec); RH (10sec); CS (15-18 secs, if actuation signal present); CC pumps (20 sec); SX pumps (25 sec); AF 1A pump (35 sec); CS pump (40 sec, if acutaion signal now present but not present at 18 sec)

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
D/G Relaying/ schematic	DG-2/ sequencing order	1	
9 EDGs and Aux sys lesson plan		7	7
Chp 20 Essential Service Water sys Lesson plan		7	8

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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The following conditions exist on Unit 1:

- Unit is in MODE 3
- A cooldown had just been initiated
- Steam Dump Bypass Interlock control switches have just been taken to BYPASS
- No other operator actions have been performed
- The Steam Dump valves fail open and the following parameters are observed:
- RCS temperature - 537°F (A); 539°F (B); 538°F (C); 538°F (D)
- Pzr pressure - 1820 psig
- Pzr level - 10%
- S/G pressure - 850 psig (A); 740 psig (B); 800 psig (C); 750 psig (D)
- S/G flow - 1.0 Mlb/hr (A); 1.5 Mlb/hr (B); 1.1 Mlb/hr (C); 1.6 Mlb/hr (D)
- The level in the RCDDT has risen to the alarm setpoint (80%) for
REACTOR COOLANT DRAIN TANK UNIT 1 LEVEL HI-LO

Assuming all systems are functioning correctly, what is the status of the RCDDT system?

- a. BOTH RCDDT pumps are running and flow is directed to the Holdup Tanks.
- b. BOTH RCDDT pumps are running and flow is recirculated back to the RCDDT.
- c. ONE RCDDT pump is running and flow is directed to the Holdup Tanks.
- d. NEITHER RCDDT pump is running and NO flow exists for the system.

Answer: d Exam Level: B Cognitive Level: Comprehension Facility: Braidwood Exam Date: 9/14/98
 KA: 068 A4.04 RO Value: 3.8 SRO Value: 3.7 Section: SYS RO Group: 1 SRO Group: 1
 Item/Evolution: Liquid Radwaste System
 NA Ability to manually operate and/or monitor in the control room:
 Automatic isolation
 Explanation of Answer: Conditions for steam flow & low RCS temp. actuate SI. The coincident CNMT Phase A Isolation signal isolates RCDDT valves out. Closure of valve RE9170 causes pumps to stop.

Reference Title/Facility	Reference Number	Section/Page	Revision	L. O.
PRT and RCDDT/schematic		RY-4	2	
Chp 48a Liquid Rad Waste lesson plan			6	11
Ch61 ESF lesson plan			5	7

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 59 CNMT Sump sources of input during normal operations

During at-power operations with systems in their normal alignment, what is a normal source of water to the Containment Floor Sump?

- a. SI Accumulator valve leakoffs.
- b. Leakoff from the #3 RCP seals.
- c. Leakoff from the reactor vessel flange.
- d. Valve packing leakage from the CVCS letdown isolation valves.

Answer b Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 068 K1.07 RO Value: 2.7 SRO Value: 2.9 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Liquid Radwaste System

KA Knowledge of the physical connections and/or cause-effect relationships between Liquid Radwaste System and the following Sources of liquid wastes for LRS

Explanation of Answer Rx cavity sump output to CNMT Floor sump, #2 seals directed to RCDT, RV flange to RCDT, valve leakoffs directed to PRT

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Chp 46a Liquid Radwaste System		6	12

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 60 Waste Gas Decay Tank Operations

When aligned for normal operation (BOP GW-1), what is the response to high pressure sensed at the in-service Gas Decay Tank?

- 1 alarm is generated that...
- a. alerts the operator to manually place a standby Gas Decay Tank in service.
- b. indicates auto swap of in-service Gas Decay Tank to selected standby Gas Decay Tank, and alerts the operator to align another standby Gas Decay Tank.
- c. indicates auto swap of in-service Gas Decay Tank to selected standby Gas Decay Tank and auto swap of standby Gas Decay Tank to new standby Gas Decay Tank.
- d. shuts down the Waste Gas Compressors and isolates the in-service Gas Decay Tank.

Answer b Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 071 A4.05 RO Value: 2.6 SRO Value: 2.6 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Waste Gas Disposal System

KA Ability to manually operate and/or monitor in the control room:
Gas decay tanks, including valves, indicators, and sample line

Explanation of Answer Indicates auto swap to standby WGD Tank at 95 psig.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Gas waste sys S/U & Operation/ BwOP GW-6		5	
GDT sel sw reposition req'd/ 0GW02J-A1		51	
Chgp 46 Gas Radwaste sys lesson plan		6	6

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 6/ Check Source operation

Area Radiation Monitor for Fuel Bldg Fuel Handling Incident (0RE-AR055) is being manually Check Source tested. What is the system response when the monitor's CHECK SOURCE (C/S) pushbutton is depressed at the RM-23 panel?

- a. The alarm and automatic action output will be blocked, and the RM-23 amber INTLK LED will be lit.
- b. The alarm and automatic action output will be blocked, and the RM-23 green AVAIL LED will be lit.
- c. The alarm will actuate when the alert setpoint value is reached, and the RM-23 amber INTLK LED will be lit.
- d. The alarm will actuate when the high setpoint value is reached, and the RM-23 red HIGH LED will be lit.

Answer b Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98
 KA: 072 A4.03 RO Value: 3.1 SRO Value: 3.1 Section: SYS RO Group: 1 SRO Group: 1
 System/Evolution Area Radiation Monitoring System
 KA Ability to manually operate and/or monitor in the control room:
 Check source for operability demonstration
 Explanation of Answer Depressing the C/S blocks the alarm and auto function of the minitor but the AVAIL litght remains lit.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Control Function Channel			
Check Source Energized/BwOP			
AR/PR-11A26	B.1	1	
Rad Monitor Sys lesson plan chp 49		7	3, 8

Material Required for Examination
 Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Question **62** Loss of FHB Overhead Crane rad monitor

The following conditions exist on Unit 2:

- Refueling operations are in progress

While using the Fuel Handling Building Crane to move new fuel into the Spent Fuel Pool, the radiation monitor 0RE-AR039, Fuel Handling Building Crane Monitor, goes into alarm. What action is affected?

- a. Traverse of the Fuel Handling Building Crane bridge and trolley.
- b. Both lowering and raising the Fuel Handling Building Crane hoist.
- c. Traverse of the Fuel Handling Building Crane trolley and raising the hoist.
- d. Raising the Fuel Handling Building Crane hoist.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 072 K3.02 RO Value: 3.1 SRO Value: 3.5 Section: SYS RO Group: 1 SRO Group: 1

System/Evolution Area Radiation Monitoring System

KA Knowledge of the effect that a loss or malfunction of the Area Radiation Monitoring System will have on the following:
Fuel handling operations

Explanation of Answer Rad monitor prevents raising hoist.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Chp 49, Radiation Monitors lesson plan		7	4.a.3)

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 63 Evaluation of eqpt affected for slow loss

The following conditions exist on Unit 1:

- A unit startup is in progress with reactor power raised above 18%.
- Turbine is at 1800 rpm ready to be synchronized to grid.
- Motor driven feedwater pump is supplying the S/Gs with Feed Reg Bypass valves in AUTO.
- Steam Dump demand in AUTO at 12%.
- Instrument air header pressure begins to slowly drop due to a leak

If the leak CANNOT be isolated and instrument air pressure continues to drop, which of the following would occur?

(Assume NO operator action taken.)

- a. AF recirculation flow to the CST would be lost due to AF recirc, 1AF022A, failing closed.
- b. Pressurizer level would increase due to charging header flow control valve, 1CV121, failing open.
- c. Pressurizer pressure would decrease due to Aux spray isolation, 1CV8145, failing open.
- d. Feedwater heater 17A extraction steam would isolate due to emergency drain, 1HD038A, failing closed.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 078 K3.02 RO Value: 3.4 SRO Value: 3.6 Section: SYS RO Group: 3 SRO Group: 3

System/Evolution Instrument Air System

KA Knowledge of the effect that a loss or malfunction of the Instrument Air System will have on the following:
Systems having pneumatic valves and controls

Explanation of answer Charging flow goes to maximum due to 1CV121 failing open, and letdown isol 1CV459 & 1CV460 fail closed. 'a' is incorrect because both 1A & 1B AF pump recirc valves fail open. 'c' main turbine not directly affected. 'd' not occur because steam dumps fail closed.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Loss of Instrument Air Lesson Plan			
1BWOA SEC-4	Table A	52	
Chp 53 IA/SA lesson plan		8	9

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question **64** Effect of loss of DC - CO2 actuation

With the fire protection systems in their normal alignment, what is the affect of a loss of DC power?

Loss of DC control power to the...

- a. halon control cabinet will cause halon release in the Upper Cable Spreading Room.
- b. battery control panel will cause automatic start of the diesel driven fire pump.
- c. fire detection system will cause start of the motor driven fire pump.
- d. carbon dioxide system will cause the master EMPC valve to open pressurizing the CO2 header.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 086 K4.06 RO Value: 3.0 SRO Value: 3.3 Section: SYS RO Group: 2 SRO Group: 2

System/Evolution Fire Protection System

KA Knowledge of Fire Protection System design feature(s) and or interlock(s) which provide for the following:
CO2

Explanation of Answer EMPCs uses DC control power. On loss of power, the master EMPC valves fail open which in turn cause the master discharge/selector valve to open, charging the affected header.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Chp 57, Fire Protection System lesson plan		5	8

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 65 Evaluate conditions - unwarranted rod withdrawal

The following conditions exist on Unit 1:

- Reactor power is 30%.
- Rod control is in Automatic
- Tref - 564°F
- Tave values - 564°F (A); 565°F (B); 565°F (C); 564°F (D)
- Power Range NI - 31% (N41); 29% (N42), 30% (N43); 30% (N44)
- Control bank D is at 156 steps.

Which condition would result in continuous rod withdrawal?

- a. Turbine first stage pressure PT-505 fails to 100%.
- b. Power Range channel N41 fails to 20%.
- c. Loop A Tcold fails 553°F.
- d. Tref signal fails 557°F.

Answer a Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 001 AA2.05 RO Value: 4.4 SRO Value: 4.6 Section: EPE RO Group: 2 SRO Group: 1

System/Evolution Continuous Rod Withdrawal

KA Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal: Uncontrolled rod withdrawal, from available indications

Explanation of Answer Input to rod control Tref, auctioneered HIGH Tave & Auctioneered high PRNIs: PT-505 provides input signal for development of Tref. If it fails high Tref goes to maximum value (581°F) and results in rods being withdrawn to match Tave to Tref. PR failure high compares the rate of change of reactor power to the rate of change of turbine power. Initially high rate of change during failure but rapidly the rate of change falls to zero and so rods may initially begin to insert but quickly stop motion with no more rate of change. Auctioneered high Tave is used and Tcold failing low will remove this input (if previously auctioneered high). Tref failing low will cause rods to move inward to match Tave to Tref.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Rod control Unit/ Schematic	RD-2	2	
Chp 28 Rod control sys Lesson Plan		12	20
Uncontrolled Rod Motion/1BwOA ROD -1			
Lesson plan		6	3

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

A Control Bank D rod was dropped from 156 steps. The P-A converter was left at 156 steps when it was to be reset to ZERO steps as directed by procedure BwOA ROD-3 "Dropped Rod Recovery".

What is the effect of performing the procedure in this manner?

- a. While performing the procedure, the C-11 Rod Stop will be received prior to realigning the rod.
- b. While performing the procedure, the Rod Insertion Limit Alarm will be received at a lower rod position than required.
- c. After the procedure is complete, Bank C control rods will begin insertion at a lower value of Control Bank D.
- d. After the procedure is complete, Bank C control rods will begin insertion at a higher value of Control Bank D.

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 003 AK3.10 RO Value: 3.2 SRO Value: 4.2 Section: EPE RO Group: 2 SRO Group: 1

System/Evolution Dropped Control Rod

KA Knowledge of the reasons for the following responses as they apply to Dropped Control Rod:
RIL and PDIL

Explanation of Answer The bank overlap units are bypassed when rods are moved with individual bank selector positions. The P to A converter provides step information to rod position indication including the C-11 circuit. As the individual rod was withdrawn to approximately 67 steps the C11 circuit would sense that bank D was at 223 steps and block outward motion.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
RD Data logging/ rod stops schematic	RD-5/RD-1 P/A & C-11 rod stop	0/0 12	
C 28 Rod Control sys lesson plan			1g, 10

Material Required for Examination

Question Source: New

Question Modification Method: Editorially Modified

Question Source Comments: D.C. Cook 6/13/1995

Comment Type Comment

Question 67 Stabilized RCS temperature with failure of Steam Dumps

On Unit 1, a loss of all circulating water pumps has resulted in a reactor trip. All control systems respond as expected. Significant decay heat causes RCS temperature to increase following the trip.

what RCS temperature should temperature stabilize?

Temperature should stabilize at...

- a. 550°F.
- b. 557°F.
- c. 561°F.
- d. 565°F.

Answer C Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 007 EA1.03 RO Value: 4.2 SRO Value: 4.1 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Reactor Trip

KA Ability to operate and / or monitor the following as they apply to Reactor Trip:
RCS pressure and temperature

Explanation of Answer The condenser would NOT be available for steam dumps (either on trip controller or load rejection controller).
The S/G pressure would stabilize based on the secondary PORV opening setpoint normally set at 1115 psig.
The Main Steam safety valve setting is 1175 psig.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Steam dumps/schematic	MS-4/ C-9	4	
Chp 24 Steam dumps lesson plan		7	4
Chp 23 Main steam lesson plan		8	3

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question **68** Reactor Trip requirements

" Unit 2 is operating at full load, which group of conditions will result in an automatic reactor trip either directly or indirectly?

- a. RCP bus frequency(Hz):56.9 (Bus 156) 57.1(Bus 157) 56.9 (Bus 158) 57.2 (Bus 159)
- b. Power range (%): 107 (N41) 108 (N42) 108 (N43) 109 (N44)
- c. PZR pressure (psig): 2375 (PT-455) 2380 (PT-456) 2385 (PT-457) 2380 (PT-458)
- d. S/G C NR level (%): 35 (LT-537) 38 (LT-538) 38 (LT-539) 37 (LT-558)

Answer a Exam Level R Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 007 EK2.03 RO Value: 3.5 SRO Value: 3.6 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Reactor Trip

KA Knowledge of the interrelations between Reactor Trip and the following:
Reactor trip status panel

Explanation of Trp condition RCP UF - 2/4 RCP buses < 57.0 Hz. Other trip setpoints: Rx power - 2/4 >109%; Pzr pressure

Answer Title: 2/4 > 2385 psig

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
ESF Setpoints/ schematic	EF-1/Rx trip	4	
2BwEP-0 Reactor Trip or SI lesson plan		3	6
Chp 60b RPS lesson plan		6	4

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: Comanche Peak 11/94

Document Type Comment

Question 69 Tail-Pipe conditions

With the RCS at normal operating pressure and temperature, what is the condition of the steam entering the PRT at normal conditions, if a PORV opens? (Assume an ideal thermodynamic process).

- a. Superheated steam at 651°F.
- b. Superheated steam at 250°F.
- c. Saturated steam-water mixture at 222°F.
- d. Saturated steam water mixture at 163°F.

Answer C Exam Level R Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 008 AK1.01 RO Value: 3.2 SRO Value: 3.7 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Pressurizer Vapor Space Accident

KA Knowledge of the operational implications of the following concepts as they apply to Pressurizer Vapor Space Accident: Thermodynamics and flow characteristics of open or leaking valves

Explanation of Answer Nominal PRT pressure 3 psig; Hg = 1154 BTU/lb. Saturation temperature 221.9°F. At NOP Pzr pressure 2235 psig with Hg = 1117.7 BTU/lb. Therefore PRT conditions are within saturation parameters.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Steam Tables			
Chp 14, Pressurizer lesson plan		9	25e

Material Required for Examination Steam Tables

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: South Texas 9/95

Comment Type Comment

Question 70 Calculation of subcooled margin on Iconics

The following conditions exist on Unit 1:

- Subcooling Margin output from the SPDS Iconics has failed
- 1C RCP and 1D RCP are running

The Unit Supervisor has asked you to determine the subcooling margin using the same valid inputs as used by SPDS.

What are the parameters used to calculate subcooling margin?

- RCS wide range pressure from loop C hot leg and core exit thermocouple temperatures.
- Pressurizer pressure and core exit thermocouple temperatures.
- RCS wide range pressure from loop A and loop C hot leg, and RCS loop A and loop C hot leg temperatures.
- Pressurizer pressure and RCS loop C and loop D hot leg temperatures.

Answer a Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 009 EA1.10 RO Value: 3.8 SRO Value: 3.9 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Small Break LOCA

Ability to operate and / or monitor the following as they apply to Small Break LOCA:
Safety parameter display system

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
SPDS Display schematic	CX-1/subcooling	1	
C.4b Inadequate Core Cooling Lesson plan		7	6

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

The following conditions exist during performance of BEP-0.

- Train A ECCS pumps failed to start.
- RCS pressure is 1350 psig.
- Containment pressure of 7 psig.
- Bus 142 has an overcurrent trip on the normal feeder breaker.
- SI actuated due to High Containment Pressure.
- The highest critical safety function is Yellow on Heat Sink.
- All other equipment and components operated as expected.

Based on above plant conditions, the RCPs should...

- a. remain running because NO SI pumps or Charging Pumps are running.
- b. be stopped because RCS pressure is below the RCP trip criteria.
- c. remain running until Pressurizer level decreases below 34%.
- d. be stopped because CC flowpath to the RCP motor oil coolers is isolated.

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 011 EA1.03 RO Value: 4.0 SRO Value: 4.0 Section: EPE RO Group: 2 SRO Group: 1

System/Evolution Large Break LOCA

KA Ability to operate and / or monitor the following as they apply to Large Break LOCA:
Securing of RCPs

Explanation of Answer The trip criteria is < 1425 psig, with NO cooldown in progress, and HHSI flow > 50 gpm or SI flow > 100 gpm.

Source Title/Facility Reference Number	Section/Page	Revision	L. O.
Course for 1BwEP-0	Trip RCPs	1C	
1BwEP-0 lesson plan	RCP trip criteria	11	2,5

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: Watts Bar 3/3/1995

Comment Type Comment

Question 72 Eval loss of cooling flow

On a loss of seal injection to the RCPs, what criteria is used to determine if the RCPs should be tripped per BWOA RCP-2 "Loss Of Seal Cooling"?

- a. High temperatures on the RCP lower bearing outlet temperatures.
- b. Time elapsed since loss of seal injection.
- c. RCP Thermal Barrier Component Cooling Water low flow alarms.
- d. High vibration condition on the RCP.

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 015 AA2.10 RO Value: 3.7 SRO Value: 3.7 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Reactor Coolant Pump Malfunctions

KA Ability to determine and interpret the following as they apply to Reactor Coolant Pump Malfunctions:
When to secure RCPs on loss of cooling or seal injection

Explanation of Answer Seal & bearing temperatures are monitored for trip setpoint.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Loss of seal cooling 1BWOA RCP-2		54	
Loss of Seal Cooling lesson plan		6	4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 73 Eval of RCP seal failure

Unit 1 is operating at 100% power when the following alarms are received/reported: .

- RCP SEAL LEAKOFF FLOW LC W (1-7-C3)

The NSO investigates and reports the following additional information:

- RCP 1A seal injection flow is 10.7 gpm
- #1 Seal Leakoff Flow on 1A RCP is 0.4 gpm
- RCP 1A Seal Water Outlet Temperature is 140°F and STABLE
- RCP 1A Bearing Outlet Temperature is 145°F and STABLE
- Unit 1 RCDT level indicates 75%

Based on the above information, which of the following events has occurred?

- a. RCP 1A #1 Seal has failed closed
- b. RCP 1A #1 Seal has failed open.
- c. RCP 1A #2 Seal has failed closed.
- d. RCP 1A #2 Seal has failed open.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 015 AK2.07 RO Value: 2.9 SRO Value: 2.9 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Reactor Coolant Pump Malfunctions

KA Knowledge of the interrelations between Reactor Coolant Pump Malfunctions and the following:
RCP seals

Explanation of answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
RCP seal Failure/ 1BwOA RCP-1		55B	
1BwOA RCP-1 lesson plan		7	5

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Editorially Modified

Question Source Comments: Braidwood bank

Comment Type Comment

Question 74 VCT level transmitter malfunction

Given the following:

- The plant is at 90% power with ALL controls in AUTO.
- VCT level transmitter, LT-112, fails HIGH causing a letdown diversion.
- At the time of failure VCT level transmitter, LT-185, reads 50%.

What will occur if NO operator action is taken?

VCT level decreases...

- a. until Auto makeup starts and maintains VCT level.
- b. with NO auto makeup capability and charging suction shifts to RWST.
- c. faster than auto makeup input and charging suction shifts to RWST.
- d. until charging pumps lose suction and start to cavitate.

Answer	d	Exam Level	B	Cognitive Level	Application	Facility:	Braidwood	ExamDate:	9/14/98
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KA:	022 AA1.08	RO Value:	3.4	SRO Value:	3.3	Section:	EPE	RO Group:	2	SRO Group:	2
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System/Evolution Loss of Reactor Coolant Makeup

KA Ability to operate and / or monitor the following as they apply to Loss of Reactor Coolant Makeup:
VCT level

Explanation of LT 112 provides for AUTO makeup to the VCT. If NO operator action taken, then level will continue to fall until
Answer NPSH is lost to the CENT CHG pump(s). Transfer will NOT occur to RWST since both channels are required

for swap. An alarm will be generated from LT-185 at 20% level.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
CVCS notes/schematic	CV-2/ LT 112 table	3	
CVCS 15a CVCS lesson plan		10	11,14

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Question 75 Time/amount E-boration for condition

Given the following after a reactor trip:

- THREE rods remain withdrawn.
- Due to equipment malfunctions boration is only available from the RWST.
- Charging flow rate 132 gpm.
- RCS boron concentration was 1050 prior to the trip.
- 120 gpm letdown in service.

Of the listed times, which would be minimum acceptable time that boration from the RWST would have to occur?

- a. 1 Hour
- b. 2 Hours
- c. 3 Hours
- d. 4 Hours

Answer b Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 024 AA2.05 RO Value: 3.3 SRO Value: 3.9 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Emergency Boration

KA Ability to determine and interpret the following as they apply to Emergency Boration:
Amount of boron to add to achieve required SDM

Explanation of Answer 1BwEP ES-0.1 requires 3600 gallons boration from RWST for each rod not fully inserted, therefore requiring 10,800 gallons. If net change over is 120 gpm, then required time is 10,800/120 = 90 minutes. Other answers based on counting 2 rods and/or borating from CV-8104 @ 57 gpm with total of 1200 gallons.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
JA Pri-2 emergency Boration		55B	
1BwOA Pri-2 lesson plan		1	4,6
1BwEP-0 lesson plan		11	3

Material Required for Examination 1BwEP ES-0.1, page 6 (step 5)
Question Source: New Question Modification Method:

Question Source Comments:
Comment Type Comment

Question 76 Calc of time to saturation/core boiling

The following conditions exist on Unit 1:

- A forced outage is in progress
- The plant was shutdown 8½ days ago to repair a steam generator tube leak.
- Draining of the RCS was initiated to allow access to S/Gs.
- Reactor vessel level is at 397' 1" with Thot at 212°F.
- A loss of RHR pumps due to cavitation has occurred

Which of the following is the smallest amount of flow that meets the minimum makeup flow required to maintain current RCS level?

- a. 80 gpm.
- b. 72 gpm.
- c. 65 gpm.
- d. 59 gpm.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 025 AK1.01 RO Value: 3.9 SRO Value: 4.3 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Loss of Residual Heat Removal System

KA Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation

Explanation of Answer 8 1/2 days is 204 afters shutdown. The curve shows minimum flow at approximately 70 gpm.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Loss of RH cooling/1BWOA Pri-10		56	
JA Pri-10 Lesson plan			4

Material Required for Examination

Figure 1BWOA PRI 10-1

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 77 Alternate RCS cooling

The following conditions exist on Unit 2:

- MODE 5 operation during normal cooldown
- RCS temperature - 195° F
- RCS pressure - 325 psig
- Train A RH in service, train B RHR tagged out for repairs

What is the preferred method of core cooling if a loss of RH cooling occurs?

Alternate RCS cooling using...

- a. the SI accumulators.
- b. the S/Gs.
- c. normal charging and RHR letdown.
- d. SI Pump hot leg injection.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 025 AK3.01 RO Value: 3.1 SRO Value: 3.4 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Loss of Residual Heat Removal System

KA Knowledge of the reasons for the following responses as they apply to Loss of Residual Heat Removal System:
Shift to alternate flowpath

Explanation of Answer Steaming Intact/non-isolated SGs is the preferred alternate decay heat removal method if the RCS is intact.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Loss of RHR Cooling/1BWOA Pri-10 1BWOAPri-10 Lesson Plan	Table A	56	4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 78 Evaluation of CCW leak

The following conditions exist on Unit 1:

- The reactor is shutdown.
- RHR is in shutdown cooling.
- RCS temperature is 300°F.
- RCS pressure is 160 psig.
- CCW surge tank level is decreasing

What leak locations will produce these indications?

- a. RHR Heat Exchanger
- b. Thermal Bearing Heat Exchanger
- c. Letdown Heat Exchanger
- d. Seal Water Heat Exchanger

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 026 AA1.05 RO Value: 3.1 SRO Value: 3.1 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Loss of Component Cooling Water

KA Ability to operate and / or monitor the following as they apply to Loss of Component Cooling Water:
The CCWS surge tank, including level control and level alarms, and radiation alarm

Explanation of The seal water HX would be the only location where the CC pressure would be lower than the process fluid
Answer pressure. RHR HX approx. 165 psig; L/D Hx pressure should be approximately 160 psig, & Thermal barrier pressure should be about 160 psig.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
CCW malfs/ 1BwOA Pri-6	Att B	56	
DA Pri-6 lesson plan	Att B	6	3

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method: Significantly Modified

Question Source Comments: Zion 7/13/92

Comment Type	Comment

Question 79 Pressure controller step change

The following conditions exist on Unit 2:

- Reactor power is 100%
- Pressurizer pressure control is in automatic.

What is the immediate response of the pressure control system if the Master Pressure Controller setpoint is inadvertently changed to 2330 psig (step change)?

- a. PORV RY455A opens and spray valves open.
- b. PORV RY455A opens, spray valves open, and all heaters energize.
- c. Spray valves open and proportional heaters go to minimum.
- d. Spray valves close and proportional heaters go to maximum.

Answer d Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98
 KA: 027 AA1.01 RO Value: 4.0 SRO Value: 3.9 Section: EPE RO Group: 1 SRO Group: 2

System/Evolution Pressurizer Pressure Control Malfunction

KA Ability to operate and / or monitor the following as they apply to Pressurizer Pressure Control Malfunction: PZR heaters, sprays, and PORVs

Explanation of Answer Setting the pot setting higher reduces the output from the controller and raises the demanded pressure setpoint. This reduction results in spray valve closure & heaters turning fully on.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Pzr Pressure Control/ schematic	RY-2/PK-455A in Auto	3	
Chp 14 Pressurizer lesson plan		9	30

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: Calvert Cliffs 11/97

Comment Type Comment

Question 80 Non-Controlling channel failure

The following conditions exist on Unit 1:

- Reactor power is 100%
- All systems are in automatic
- Pressurizer pressure channels PT-456 and PT-458 reads normal
- Channel I Pressurizer Pressure Channel (PT-455) was declared inoperable and taken out of service with the appropriate bistables placed in the tripped condition .
- Controlling pressurizer pressure channel (PT-457) fails high

Assuming NO operator action, what is the plant response to the channel failure?

- a. Both PORVs and both spray valves open resulting in a reactor trip from low pressurizer pressure followed by SI actuation.
- b. The reactor will trip on high pressure, and safety injection will actuate on low pressure due to spray valve operation.
- c. Pressurizer proportional heaters will de-energize and spray valves will open resulting in an OTdT runback prior to reactor tripping, and SI will actuate due to low pressurizer pressure.
- d. Both PORVs and both spray valves remain closed while pressurizer heaters de-energize.

Answer b Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 027 AA2.15 RO Value: 3.7 SRO Value: 4.0 Section: EPE RO Group: 1 SRO Group: 2

System/Evolution Pressurizer Pressure Control Malfunction

KA Ability to determine and interpret the following as they apply to Pressurizer Pressure Control Malfunction:
Actions to be taken if PZR pressure instrument fails high

Explanation of Answer TWO PZR pressure channels will have HIGH PZR PRESSURE bistables actuated resulting in the reactor trip. The sprays will have modulated fully open resulting in actual pressure decreasing (PORV 1RY455A would have also opened on the failure of PT-457, but would close when the PZR pressure fell to 2185 psig PT-458 will actuate the low pressure interlock closing the PORV) until SI occurs at 1829 psig.

Reference Title/Facility Reference Number	Section/Page	Revision	L. C.
Pzr Pressure Control/ schematic	RY-2/ PZR press	3	
Chp 14 Pressurizer lesson plan		9	30

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: BV 8/91

Comment Type Comment

Question 8/ Failed level channel low.

The plant is operating at 100% power with all control systems in AUTO. The following parameters are noted:

- Letdown Hx outlet flow (FI-132) - 75 gpm
- Charging Header flow (FI-121) - 87 gpm
- Total seal injection flow (FI-142 -FI -45) - 33 gpm

What is the effect on total seal injection flow initially if controlling Pzr level channel LT-459 fails LOW?

Total seal injection flow will...

- a. decrease to 0 gpm.
- b. decrease to approximately 20 gpm.
- c. remain approximately 33 gpm.
- d. increase to greater than 40 gpm.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 028 AK3.05 RO Value: 3.7 SRO Value: 4.1 Section: EPE RO Group: 3 SRO Group: 3

System/Evolution Pressurizer Level Control Malfunction

KA Knowledge of the reasons for the following responses as they apply to Pressurizer Level Control Malfunction:
Actions contained in EOP for PZR level malfunction

Explanation of Answer The failure of the level instrument low increases charging flow and charging discharge header pressure. Since seal injection flow is normally increased by throttling close on CV182 to increase backpressure, the result is the same and seal injection flow will increase.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
CVCS notes/schematic	CV-2/cvcs ratings	2	
1BwOA Inst 2 Att C lesson plan		9	1

Material Required for Examination

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

Question Source Comments: Braidwood 1996 NRC exam. Modified premise from failed controller to failed level channel. Changed location of correct answer based on different response (increasing flow instead of decreasing flow).

Comment Type Comment

Question 82 AMS conditions

The following conditions exist on Unit 1:

- At t= 0 sec, Turbine load was decreased below 352 MW (30% power)
- At t=240 sec, The running main feedwater pump tripped.
The reactor did NOT trip due equipment malfunction.
- At t=250 sec, All feedflow indications decrease to 0% flow
- At t=320 sec, All steam generator levels decrease below 15%.

Based on this information, AMS would...

- a. initiate at t=320 sec.
- b. initiate at t=345 sec.
- c. initiate at t=360 sec.
- d. NOT initiate because C-20 is cleared.

Answer	b	Exam Level:	B	Cognitive Level	Application	Facility:	Braidwood	ExamDate:	9/14/96
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KA:	2.4.48	RO Value:	3.5	SRO Value:	3.8	Section:	EPE	RO Group:	2	SRO Group:	1
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System/Evolution Anticipated Transient Without Scram

KA

Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.

Explanation of Answer AMS remains armed for 6 minutes(360 sec) following decrease below 30%(C-20). The actuation signal is generated after 3/4 SGs level have fallen 3% below the LO-2 (reactor trip) setpoints of 18% for a period of 25 seconds. C-20 would clear @ t=360sec. AMS actuation occurs at 320 + 25 = 345 sec.

Reference Title/Facility Reference Number

Section/Page

Revision

L. O.

/ / schematic

PN-3/ logic 1 schem

2

Chp 60b

6

7

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Question 83 Evaluation of SR NIS voltage failure

The following conditions exist on Unit 1:

- Reactor startup in progress
- Intermediate power range indication: 2.5E-5 amp N35 & 2.8E-5 amp N36
- SOURCE RANGE PERMISSIVE P-6 permissive light clear
- Source Range Channel N31 high voltage power supply fails to HALF its normal value

What indication(s) would be available to alert the operator to this failure?

- a. None, until power is lowered below the P-6 setpoint, and then the Source Range N31 indication will indicate lower than expected.
- b. None, until power is lowered below the P-6 setpoint, and then the Source Range N31 indication will indicate higher than expected.
- c. Annunciator SR HIGH VOLTAGE FAILURE (1-10-B1) will remain in alarm when power exceeds P-10.
- d. Annunciator SR HIGH VOLTAGE FAILURE (1-10-B1) will re-flash when the voltage source fails.

Answer a Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 032 AK1.01 RO Value: 2.5 SRO Value: 3.1 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Loss of Source Range Nuclear Instrumentation

KA Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation: Effects of voltage changes on performance

Explanation of Based on Gas filled detector curve (Region III), the number of events collected would drop (counts drop).

Answer Alarm and voltage input to SR detector is blocked until both IR NIS fall below the P-6 setpoint.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
SR High Volt Failure/ 1BWAR 1-10-B1	setpts/notes	1	
Source Range detector/schematic	NI-4	4	
Chp 31 source range nuclear inst			
Lesson plan		6	2,3,11,12

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 84 Eval of failed IR channel on SU

The following conditions exists on Unit 2:

- Plant shutdown is in progress.
- Power range channels indicate: 9% (N41), 10% (N42), 11% (N43) , 11% (N44)
- Intermediate range channel N-36 fails HIGH.

When this failure occurs, what is the plant response this failure?

- The reactor will trip on high IR flux, and source range trip will reinstate when N-35 decreases below P-6.
- The reactor will trip on high IR flux, and source range trip will NOT be automatically reinstated.
- The reactor will NOT trip immediately, but will trip when the source range trip is reinstated when N-35 decreases below P-6
- The reactor will NOT trip, and source range trip will NOT be automatically reinstated.

Answer d Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 033 AA2.04 RO Value: 3.2 SRO Value: 3.6 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Loss of Intermediate Range Nuclear Instrumentation

KA Ability to determine and interpret the following as they apply to Loss of Intermediate Range Nuclear Instrumentation: Satisfactory overlap between source-range, intermediate-range and power-range instrumentation

Explanation of Answer Since reactor power is < P-10 setpoint (10% power), the IR trip setpoint at 25% EICA will be exceeded resulting in reactor trip. SR will NOT be reinstated automatically because only one IR channel will fall below P-6 and Two are required to remove P-6.

Reference Title/Facility	Reference Number	Section/Page	Revision	L. O.
Intermediate Range/ schematic		NI-3	4	
Loss of Intermediate range nuclear inst			6	4,8,9,10
Lesson plan				

Material Required for Examination

Question Source: New Question Modification Method: Significantly Modified

Question Source Comments: Watts Bar 8/94

Comment Type Comment

Question 85 Monitors for S/G Tube leakage

The following conditions exist on Unit 1:

- Reactor power is 75%
- Troubleshooting has commenced due to reduced condenser vacuum with the air ejectors out of service.
- Hogging vacuum pumps are aligned to the main condenser to aid in maintaining vacuum.

What would **NOT** be an indication of a Steam Generator Tube Leak under these conditions?

- a. Increasing conductivity levels for the main condenser hotwell.
- b. Increasing radiation level on 1RE-PR027, "S/JAE/Gland Steam Exhaust Monitor".
- c. Decreasing feed flow to ONE S/G.
- d. Increasing radiation levels on 1RE-PR08 "S/G Blowdown Monitor".

Answer a Exam Level R Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 037 AA1.02 RO Value: 3.1 SRO Value: 2.9 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Steam Generator Tube Leak

KA Ability to operate and / or monitor the following as they apply to Steam Generator Tube Leak:
Condensate exhaust system

Explanation of Answer The Hogger discharge is aligned through the Off Gas header which is monitored by 1RE-PR027.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
SGTR lesson plan/ Bw0A Sec 8		6	4
Ch 49 rad monitors lesson plan		7	14

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 86 Loss of subcooling

Lesson 3 "Steam Generator Tube Rupture" is being performed in response to a tube rupture on 2C S/G. The cooldown has just been completed but the target temperature value selected by the operators was higher than that stipulated in the procedure.

What condition could result because of this error?

- a. Loss of RCS subcooling before RCS and ruptured S/G pressures are equalized.
- b. Increase in pressure of the ruptured S/G with resultant lifting of the S/G Safety Valve.
- c. Increase in pressure of the non-ruptured S/Gs with resultant lifting of their S/G Safety Valves.
- d. Filling the Pressurizer solid during the subsequent depressurization.

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 038 EK3.06 RO Value: 4.2 SRO Value: 4.5 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Steam Generator Tube Rupture

KA Knowledge of the reasons for the following responses as they apply to Steam Generator Tube Rupture: Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
SGTR lesson plan 1BwEP-3		12	1
ERG basis			

Material Required for Examination

Question Source: New Question Modification Method: Editorially Modified

Question Source Comments: Salem 6/94

Comment Type Comment

Question 87 Steamline isolation

The following conditions exist on Unit 1:

- The Unit was in MODE 3 at normal operating temperature and pressure prior to the event.
- A faulted steam generator has occurred.
- RCS hot leg temperatures - 547°F (A), 544°F (B), 545°F (C), 547°F (D)
- RCS cold leg temperatures - 545°F (A), 530°F (B), 543°F (C), 545°F (D)
- S/G pressures - 700 psig (A), 635 psig (B), 690 psig (C), 705 psig (D)
- S/G flow - 0.85 MLB/hr (B)
- Containment pressure (Channel) - 8 psig (1), 7.5 psig (2), 7.5 psig (3), 8 psig (4)

Based on these conditions, a main steam line isolation should...

- a. have occurred because of the low pressure in at least ONE S/G.
- b. have occurred because the steamline high negative rate occurred in S/G 1B.
- c. NOT have occurred because Containment pressure is below the setpoint for the CNMT High-2 pressure signal.
- d. NOT have occurred because THREE S/Gs have pressures above the isolation setpoint and do NOT indicate high steam flow.

Answer	a	Exam Level	B	Cognitive Level	Application	Facility:	Braidwood	ExamDate:	9/14/98
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KA:	040 AA1.01	RO Value:	4.6	SRO Value:	4.6	Section:	EPE	RO Group:	1	SRO Group:	1
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System/Evolution Steam Line Rupture

KA Ability to operate and / or monitor the following as they apply to Steam Line Rupture:
Manual and automatic ESFAS initiation

Explanation of answer	The steamline isolation signal is generated by the low pressure sensed on 2/3 pressure transmitters in any one SG. CNMT pressure is below the MSLI setpoint of 8.2 psig and steamline negative rate is blocked since initial condition has PZR pressure > P-11.
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Reference Title/Facility	Reference Number	Section/Page	Revision	L. O.
ESF Setpoints/schematic		EF-2/ Stmline isol	5	
Ch 23 Main steam Sys lesson plan			8	5,13,15,16
Ch 61 ESF lesson plan			5	7

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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The following conditions exist on Unit 1 following a trip from 100% power:

- Pressurizer level is 0%
- Pressurizer pressure is 1500 psig
- Containment Pressure is 16 psig.
- Tcold is 420°F for all loops.

Where is the location of the leak?

- a. On one loop RCS cold leg.
- b. On a Main Steam Line inside containment.
- c. In a Steam Generator Tube.
- d. On a feedwater line between FWRV and Associated FWIV, 1FW009.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 040 AK1.06 RO Value: 3.7 SRO Value: 3.8 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Steam Line Rupture

KA Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture:
High-energy steam line break considerations

Explanation of Answer Secondary LOCA not indicated since Tcold is the same in all loops and RCS tcold is not consistent with given CNMT pressure for steam/feed break. SGTR not indicated since CNMT pressure is elevated. LOCA condition indicated by consistent Tcold, and CNMT pressure increase.

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
1BwEP-0 Reactor Trip or SI lesson plan		3	6,7
1RwEP2 Faulted S?g isolation lesson plan		7	2,4

Material Required for Examination

Question Source: New

Question Modification Method: Editorially Modified

Question Source Comments: St. Lucie 10/13/97

Comment Type Comment

Question 89 Eval of conditions

In accordance with BWOA SEC-3, "Loss of Condenser Vacuum", which of the following sets of conditions requires the operator to trip the reactor?

- a. LOW POWER TRIP BLOCKED P-8 annunciator - LIT
Turbine load - 200 MW
Condenser pressure - 5.2 " HgA
- b. LOW POWER TRIP BLOCKED P-8 annunciator - LIT
Turbine load - 300 MW
Condenser pressure - 6.3" HgA
- c. LOW POWER TRIP BLOCKED P-8 annunciator - CLEAR
Turbine load - 600 MW
Condenser pressure - 7.2" HgA
- d. LOW POWER TRIP BLOCKED P-8 annunciator - CLEAR
Turbine load - 900 MW
Condenser pressure - 7.8" HgA

Answer b Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98

KA: 051 AA2.02 RO Value: 3.9 SRO Value: 4.1 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Loss of Condenser Vacuum

KA Ability to determine and interpret the following as they apply to Loss of Condenser Vacuum:
Conditions requiring reactor and/or turbine trip

Explanation of Answer P-8 permissive active below 30% power (annunciator lit). At 480 MW and below, the minimum acceptable condenser pressure is 5.5 in HgA. At 600 MW minimum acceptable pressure is 7.8 in HgA. At 610 MW and greater, minimum acceptable pressure is 8.0 in HG

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
1. BWOA Sec -3 loss of condenser vac lesson plan		6	5

Material Required for Examination

Figure 1BWOA SEC 3-1

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 90 Identification of RCP seal LOCA/cooldown

Select the primary reason for rapidly depressurizing the steam generators during a Loss of All AC.

- a. To provide maximum core cooling until power can be restored.
- b. To minimize RCS inventory loss from RCP seals.
- c. To enhance restoration of S/G level from the diesel driven AF pump.
- d. To increase subcooling of the RCS.

Answer b Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 055 EK3.02 RO Value: 4.3 SRO Value: 4.6 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Station Blackout

KA Knowledge of the reasons for the following responses as they apply to Station Blackout:
Actions contained in EOP for loss of offsite and onsite power

Explanation of Answer The rapid cooling allows depressuring the RCS reducing the leak rate via the RCP seals

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Loss of All AC Power/ 1BwCA 0.0	Caution 2	1B Wog 1B	
1BwCA 0.0 lesson plan		4	

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Question 9/ Reset of sequencer

How would the sequencer operate if a Safety Injection (SI) actuation occurs while the sequencer is sequencing loads in response to an ESF bus undervoltage condition?

- a. There will be no change in operation; the undervoltage sequence overrides the SI sequence.
- b. The undervoltage sequencing stops, the sequencer immediately resets and SI loads NOT already running will sequentially start.
- c. The undervoltage sequencing stops, all started loads are shed, and SI loads will sequentially start.
- d. The undervoltage sequencing completes its cycle, then resets to SI mode, and SI loads NOT already running will sequentially start.

Answer b Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 056 AA1.21 RO Value: 3.3 SRO Value: 3.3 Section: EPE RO Group: 3 SRO Group: 3

System/Evolution Loss of Off-Site Power

KA Ability to operate and / or monitor the following as they apply to Loss of Off-Site Power:
Reset of the ESF load sequencers

Explanation of Answer The UV sequence is stopped and the SARA sequencing is initiated from step 1.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
D/G Relaying schematic	DG-2/ SARA & SDRA	1	
Ch 9 EDG and Aux sys lesson plan		7	7
Ch 4 AC Electrical distribution lesson plan		8	10,16
Ch 61 ESF lesson plan		5	7,8

Material Required for Examination

Question Source: New

Question Modification Method: Significantly Modified

Question Source Comments: Vogtle - 5/91

Comment Type Comment

Question 92 Eval of electric bus status

The following conditions exist on Unit 1:

- Bus 141 is powered from its normal source
- D/G 1A surveillance is being performed with the D/G paralleled to the bus

What would occur if a failure of the undervoltage relay results in a sensed undervoltage condition on Bus 141?

- a. SAT feeder breaker ACB 1412 and D/G feeder breaker ACB 1413 remain closed. The Safe Shutdown loads will NOT sequence and CANNOT be manually started from the control room.
- b. SAT feeder breaker ACB 1412 and D/G feeder breaker ACB 1413 will open. After a 10-second delay, ACB 1413 will close and the Safe Shutdown loads will sequence.
- c. SAT feeder breaker ACB 1412 will open but D/G feeder breaker ACB 1413 will remain closed. The Safe Shutdown loads will sequence normally.
- d. SAT feeder breaker ACB 1412 will open but D/G feeder breaker ACB 1413 will remain closed. The Safe Shutdown loads will NOT sequence and CANNOT be manually started from the control room.

Answer d Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: 056 AA2.46 RO Value: 4.2 SRO Value: 4.4 Section: EPE RO Group: 3 SRO Group: 3
 System/Evolution Loss of Off-Site Power

Ability to determine and interpret the following as they apply to Loss of Off-Site Power:
That the ED/Gs have started automatically and that the bus tie breakers are closed

Explanation of Answer On sensed UV, the SAT feeder breaker opens (and alternate feeder breaker would also have opened if closed) and the control switches for the safe shutdown loads will be locked out.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Ch 4 AC Electrical Distribution		8	10,16

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment
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Question 93 Eqpt affected on bus loss

On Unit 1 power is lost to 120 VAC Instrument Bus 111

How are the ESF and Safe Shutdown loads affected?

- a. "A" Train ESF loads will NOT load on an SI signal, but Safe Shutdown loads will load on a U/V signal.
"B" Train loads are NOT affected.
- b. A" Train ESF loads will load on an SI signal, but Safe Shutdown loads will NOT load on a U/V signal.
"B" Train loads are NOT affected.
- c. "A" Train ESF loads will NOT load on an SI signal, and Safe Shutdown loads will NOT load on a U/V signal.
"B" Train loads are NOT affected.
- d. "A" Train AND "B" Train ESF loads will NOT load on an SI signal, but Safe Shutdown loads will load on a U/V signal.

Answer C Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98

KA: 057 AA2.19 RO Value: 4.0 SRO Value: 4.3 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Loss of Vital AC Instrument Bus

KA Ability to determine and interpret the following as they apply to Loss of Vital AC Instrument Bus:
The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
0A Elec 2 Loss of inst bus	Table A	7	
00a SSPS lesson plan		3	11
1BWOA elec 2 lesson plan		6	3,5
I and C system notes	I & C 1		

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question **94** Operations required for transfer

Which of the following sets of indications are available on the Remote Shutdown Panel?

- a. Emergency boration flow, S/G level, and RCS wide range temperature.
- b. Red and green lights for reactor trip breaker position indication, S/G pressure, and pressurizer level.
- c. Main feedwater flow, letdown flow, and charging line pressure.
- d. Containment pressure, charging flow, and auxiliary feedwater flow.

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

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

068 Control Room Evacuation

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

AA1.12 Auxiliary shutdown panel controls and indicators 4.4 4.4

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
RSP PL04/5J/ schematic	PN-1	2	
Control Room Inaccessibility 1BwOA Pri-5 lesson plan	Att. A	57B	
Ch 62 Remote shutdown Panel Lesson plan		3	3,4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question **95** Major action categories

When inadequate core cooling exists, which of the following sets of actions states the proper sequence of the major action categories to be performed in accordance with BwFR-C.1, "RESPONSE TO INADEQUATE CORE COOLING", for removing decay heat from the core?

- a. Rapid secondary depressurization; reinitiation of safety injection; RCP restart.
- b. Reinitiation of safety injection; rapid secondary depressurization; RCP restart.
- c. Rapid secondary depressurization; RCP restart ; reinitiation of safety injection.
- d. RCP restart; rapid secondary depressurization; reinitiation of safety injection.

Answer **b** Exam Level **B** Cognitive Level **Comprehension** Facility: **Braidwood** ExamDate: **9/14/98**

KA: **074 EK1.03** RO Value: **4.5** SRO Value: **4.9** Section: **EPE** RO Group: **1** SRO Group: **1**

System/Evolution **Inadequate Core Cooling**

KA Knowledge of the operational implications of the following concepts as they apply to inadequate Core Cooling:
Processes for removing decay heat from the core

Explanation of
Answer

Reference Title/Facility Reference Number	Section/Page	Revisio	L. O.
Function Restoration Procedures BwFR-C.1, C.2, C.3 lesson plan		5	2,3

Material Required for Examination

Question Source: **New** Question Modification Method: **Editorially Modified**

Question Source Comments: **VC Summer 5/94**

Comment Type **Comment**

Question 96 Actions for reducing activity

High coolant activity has been detected and chemistry has determined that it is due to corrosion product activation.

Identify the effect of placing the cation demineralizer in service.

The cation demineralizer...

- a. will remove lithium so it should NOT be used in this condition.
- b. will cause the activity level to decrease as soon as it is placed in service.
- c. is NOT effective in removing corrosion product activity.
- d. is less effective than the mixed bed demineralizer so it is placed in service ONLY if decontamination factor is less than 10.

Answer b Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: 076 AA2.02 RO Value: 2.8 SRO Value: 3.4 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution High Reactor Coolant Activity

KA Ability to determine and interpret the following as they apply to High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS

Explanation of Answer The cation demin is highly effective in removing corrosion products from the coolant.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
BwOP CV-8			
1BwOA Pri-4 High coolant Activity lesson plan		1	4,5
ch 15a CVCS lesson plan		10	4

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 97 Interlocks affecting reestablishment of feed

The following conditions exist on Unit 2:

- Reactor power was 8% prior to the event below.
- A failure in the feedwater control system caused ONE S/G level to rise to 83%.
- The main turbine tripped.
- S/G levels have returned to their normal level range
- The Startup FW Pump is running

What are all the conditions that would have to be met to feed the S/Gs using the FW034's Feedwater Tempering Flow Control valves?

- a. The FW Isolation Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.
- b. The reactor trip breakers would have to be cycled, the FW Isolation Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.
- c. The FW Isolation Main Relays and Aux Relays would have to be reset and FW035 Feedwater Tempering Isol valves opened.
- d. The reactor trip breakers would have to be cycled and FW Isolation Main Relays and Aux Relays reset and FW035 Feedwater Tempering Isol valves opened.

Answer a Exam Level B Cognitive Level Application Facility: Braidwood ExamDate: 9/14/98
 KA: E05 EK2.1 RO Value: 3.7 SRO Value: 3.9 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Loss of Secondary Heat Sink

KA Knowledge of the interrelations between Loss of Secondary Heat Sink and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Explanation of Answer The P-14 signal, once clear, only maintains FWI signal via the FW Isol Aux relays if NO reactor trip signal is present. So resetting the FW Isolation Aux relay allows opening of FW035s (normal feed path at low power) and throttling of FW034s

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
ESF setpoints/ schematic	EF-2/ reset FWI	5	
Feedwater Simple/SGWLC	FW-1,2/ reset FWI	0	
Ch 61 ESF lesson plan		6	4,7,8

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Topic

Question 98 Identification of heat removal process

The following conditions exist on Unit 1:

- A leak developed on the RCS loop C flow instrument piping.
- Coincident with the RCS leak, on the reactor trip a S/G PORV failed open and was later isolated.
- FR-P.1 was entered to due to an ORANGE PATH condition.
- SI actuated and has been reset.
- All RCPs are stopped.
- Conditions required to support an RCP start are met.

Under the current conditions starting the RCP will...

- a. cause excessive thermal stresses in the stagnant loops.
- b. cause a pressure surge that will aggravate the PTS condition.
- c. provide mixing of the ECCS injection flow thereby decreasing the likelihood of PTS.
- d. increase the RCS cooldown rate thereby increase the likelihood of PTS.

Answer C Exam Level B Cognitive Level Comprehension Facility: Braidwood ExamDate: 9/14/98
 KA: E08 EK2.2 RO Value: 3.6 SRO Value: 4.0 Section: EPE RO Group: 1 SRO Group: 1

System/Evolution Title: Pressurized Thermal Shock

KA Statement: Knowledge of the interrelations between Pressurized Thermal Shock and the following:
 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.

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revisin	L. O.
FRP 1BwFR P.1, 2, lesson plan		4	3,4
Status Trees	ST-I/ Integrity		

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

Question 99 Natural Circ conditions and limits

The following conditions exist on Unit 1:

- A natural circulation is in progress per BwEP ES-0.2 "Natural Circulation Cooldown"
- Pressurizer pressure is being controlled using Aux. Spray and Pzr heaters
- As pressure is being lowered through 1300 psig, a rapid increase is noted in Pzr level
- Charging and letdown are in manual and are balanced

What actions are required to be taken by the operators?

- a. Repressurize the RCS.
- b. Isolate the SI Accumulators.
- c. Increase the RCS cooldown rate.
- d. Place excess letdown in service.

Answer a Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

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

E09 Natural Circulation Operations

EK3. Knowledge of the reasons for the following responses as they apply to Natural Circulation Operations:

EK3.1 Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics. 3.3 3.6

Explanation of Answer

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
1BwEP -0 Reactor Trip or SI Lesson plan		11	3,4,6

Material Required for Examination

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type Comment

Question 100 Reason for rapid S/G depressurization

Why are the S/Gs depressurized to less than 670 psig according to BwCA-1.1, "Loss of Emergency Coolant Recirculation"?

- a. To allow maximum AFW flow to the S/Gs.
- b. To ensure adequate subcooling for restart of the RCPs.
- c. To set up conditions for controlled injection to the RCS from the accumulators.
- d. To decrease RCS temperature and pressure which reduces break flow in a LOCA condition.

Answer C Exam Level B Cognitive Level Memory Facility: Braidwood ExamDate: 9/14/98

KA: E11 EA1.1 RO Value: 3.9 SRO Value: 4.0 Section: EPE RO Group: 2 SRO Group: 2

System/Evolution Loss of Emergency Coolant Recirculation

KA Ability to operate and / or monitor the following as they apply to Loss of Emergency Coolant Recirculation: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Explanation of Answer The concern is maximizing cooling volumes that supply water to RCS. By cooling RCS, depressurization of RCS can be initiated (while maintaining subcooling) to the point where the SI accumulators inject their volumes into the RCS.

Reference Title/Facility Reference Number	Section/Page	Revision	L. O.
Loss of Emergency Coolant Recirc/ 1BwCA-1.1		1B WOG 1B	
1BwCA 1.1 and 1.2 lesson plan		7	3

Material Required for Examination

Question Source: New

Question Modification Method: Editorially Modified

Question Source Comments: South Texas 9/92

Comment Type Comment

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

EQUATIONS

$$\dot{Q} = \dot{m}C_p\Delta T$$

$$\dot{Q} = \dot{m}\Delta h$$

$$\dot{Q} = UA\Delta T$$

$$\dot{Q} \propto \dot{m}_{\text{Net Circ}}^3$$

$$\Delta T \propto \dot{m}_{\text{Net Circ}}^2$$

$$K_{\text{eff}} = 1/(1 - \rho)$$

$$\rho = (K_{\text{eff}} - 1)/K_{\text{eff}}$$

$$\text{SUR} = 26.06/\tau$$

$$\tau = \frac{\bar{\beta} - \rho}{\lambda_{\text{eff}} \rho}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\bar{\beta}}{1 + \lambda_{\text{eff}}\tau}$$

$$\ell^* = 1 \times 10^{-4} \text{ seconds}$$

$$\lambda_{\text{eff}} = 0.1 \text{ seconds}^{-1}$$

$$P = P_0 10^{\text{SUR}(t)}$$

$$P = P_0 e^{(t/\tau)}$$

$$A = A_0 e^{-\lambda t}$$

$$CR_{S/D} = S/(1 - K_{\text{eff}})$$

$$CR_1(1 - K_{\text{eff}1}) = CR_2(1 - K_{\text{eff}2})$$

$$1/M = CR_1/CR_X$$

$$\text{DRW} \propto \phi_{\text{tip}}^2 / \phi_{\text{avg}}^2$$

$$F = PA$$

$$\dot{m} = \rho A \bar{V}$$

$$\dot{W}_{\text{Pump}} = \dot{m}\Delta P_U$$

$$E = IR$$

$$\text{Eff.} = \text{Net Work Out/Energy In}$$

$$v(P_2 - P_1) + \frac{(\bar{V}_2^2 - \bar{V}_1^2)}{2g_c} + \frac{g(z_2 - z_1)}{g_c} = 0$$

$$g_c = 32.2 \text{ lbm-ft/lbf-sec}^2$$

CONVERSIONS

$$1 \text{ Mw} = 3.41 \times 10^6 \text{ Btu/hr}$$

$$1 \text{ Curie} = 3.7 \times 10^{10} \text{ dps}$$

$$1 \text{ hp} = 2.54 \times 10^3 \text{ Btu/hr}$$

$$1 \text{ kg} = 2.21 \text{ lbm}$$

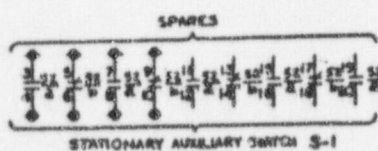
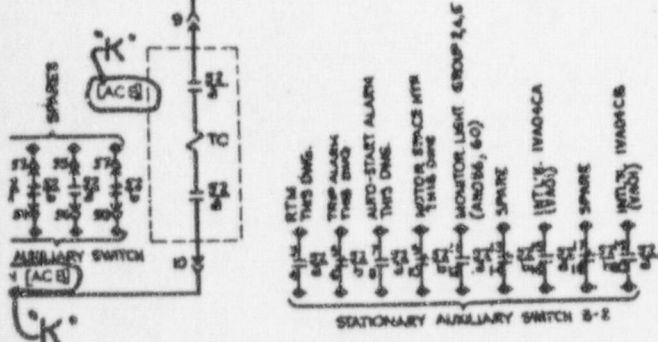
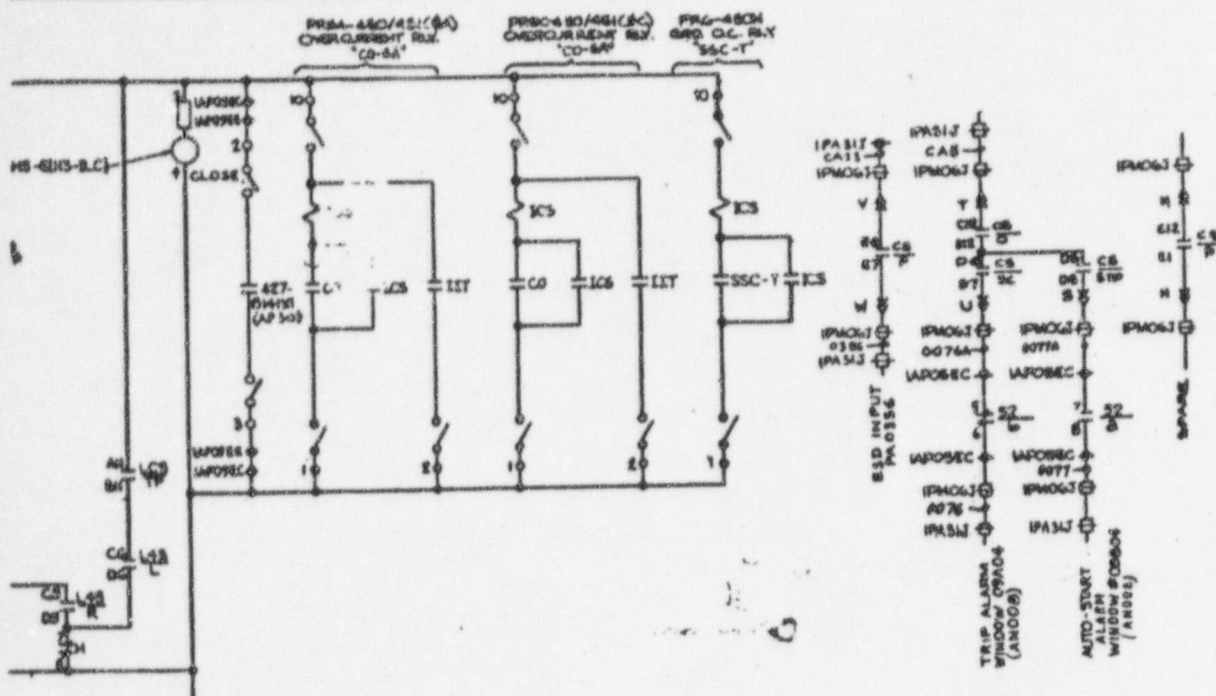
$$1 \text{ Btu} = 778 \text{ ft-lbf}$$

$$1 \text{ gal}_{\text{water}} = 8.35 \text{ lbm}$$

$$^{\circ}\text{C} = (5/9)(^{\circ}\text{F} - 32)$$

$$1 \text{ ft}^3_{\text{water}} = 7.48 \text{ gal}$$

$$^{\circ}\text{F} = (9/5)(^{\circ}\text{C}) + 32$$



RTM THIS DNG.
 B21 TRIP ALARM THIS DNG.
 B22 TRIP ALARM THIS DNG.
 B23 AUTO-START ALARM THIS DNG.
 B24 MOTOR STOPPS WITH THIS DNG.
 B25 MONITOR LIGHT GROUP 2&4 (AND 56, 60)
 B26 SPARE
 B27 INTX- I/VAO4CA
 B28 INTX- I/VAO4CA
 B29 SPARE
 B30 INTX- I/VAO4CB
 B31 INTX- I/VAO4CB



NUCLEAR SAFETY RELATED EQUIPMENT IS SHOWN ON THIS DRAWING.
 REF. N.E.CO. DLR 271C420 SHY 26

SCHEMATIC DIAGRAM
 SAFETY INJECTION PUMP 1A -
 15101PA
BRADWOOD/BRADWOOD STATION -- UNIT 1
COMMONWEALTH EDISON CO.
CHICAGO, ILLINOIS

REFERENCE DWGS.		DRAWING RELEASE RECORD		DRAWN	
DWG. NO.	DESCRIPTION	REV.	DATE	DESCRIPTION	CHKD.
E-4024 SERIES	INT/EXT W/D MCC ZAMB SAFETY FEATURES IPMO6J	K	8-2-58	ADDED CONTROL POINT FOR SAFETY FEATURES	
E-4122 SERIES	INT/EXT W/D ESF SEQUENCING AND ACTUATION CAB-TRAIN A IPB132				
E-4156 SERIES	INT/EXT W/D ANNUNCIATOR INPUT CAB: (ESF II) IPB132				
E-4611 SERIES	INT/EXT W/D 480V ESF SWGR BUS W/ IAP056				
E-4630 SERIES	S/D NOTES, LEGENDS, REFERENCE DEAN REC				
E-4661 SERIES	INT/EXT W/D 480V MCC 13K1 (IAP14)				
E-4665 SERIES	INT/EXT W/D 480V MCC 13K5 (IAP22)	K	7-27-58	ADDED CONTROL POINT FOR SAFETY FEATURES	

DATE: NONE
 CHECKED: [Signature]
 APPROVED: [Signature]
 DATE: 8-2-58

SARGENT & LUNDY

DRAWING NO. 4687
 REV. 00
 PROJECT NO. 58-1-4090 5101

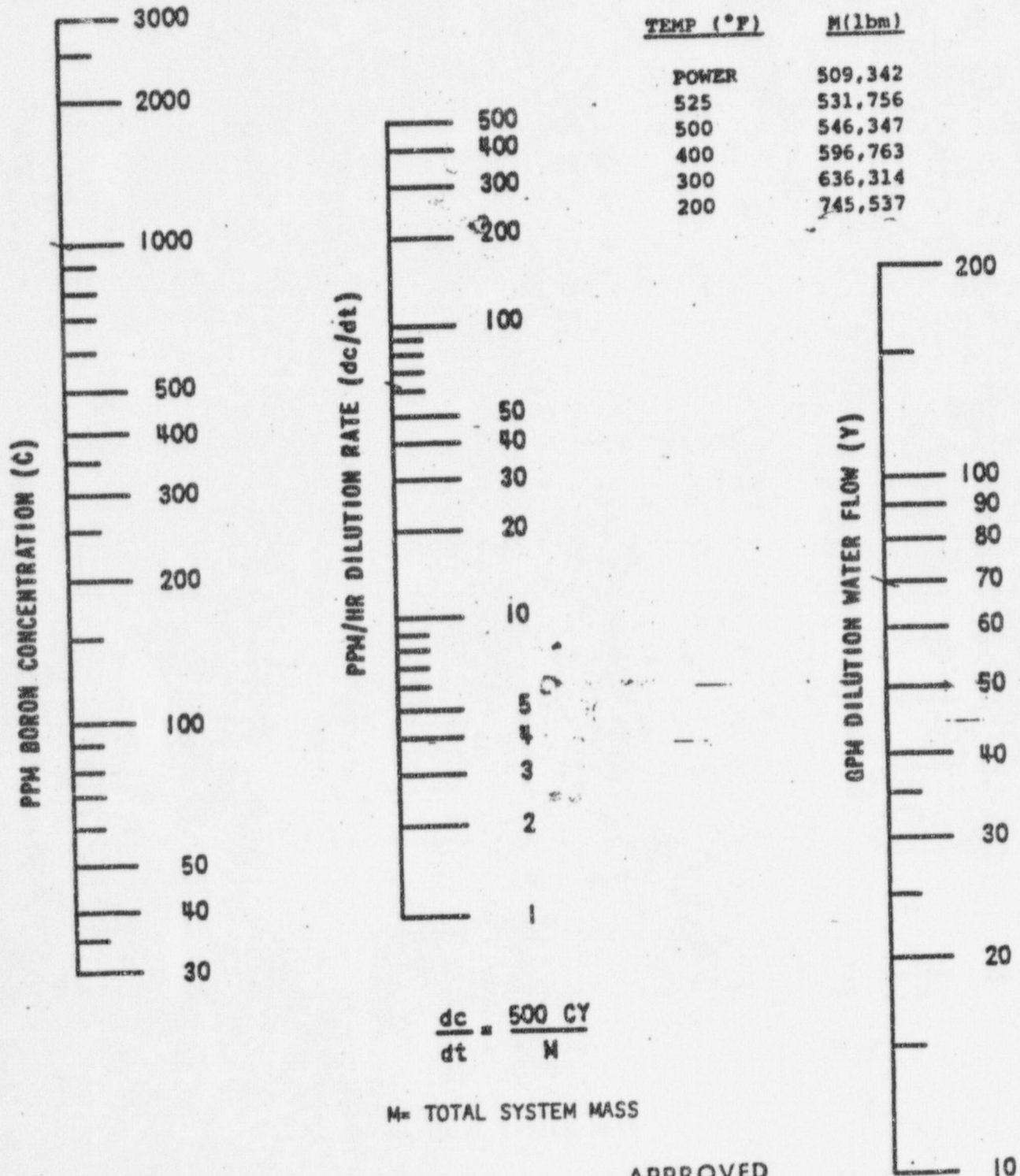
REV-9184(K) DRG-V44(K) 07

CRITICAL CONTROL ROOM DRAWING

TMCT

MULTIPLE USE

BORON DILUTION RATE NOMOGRAPH



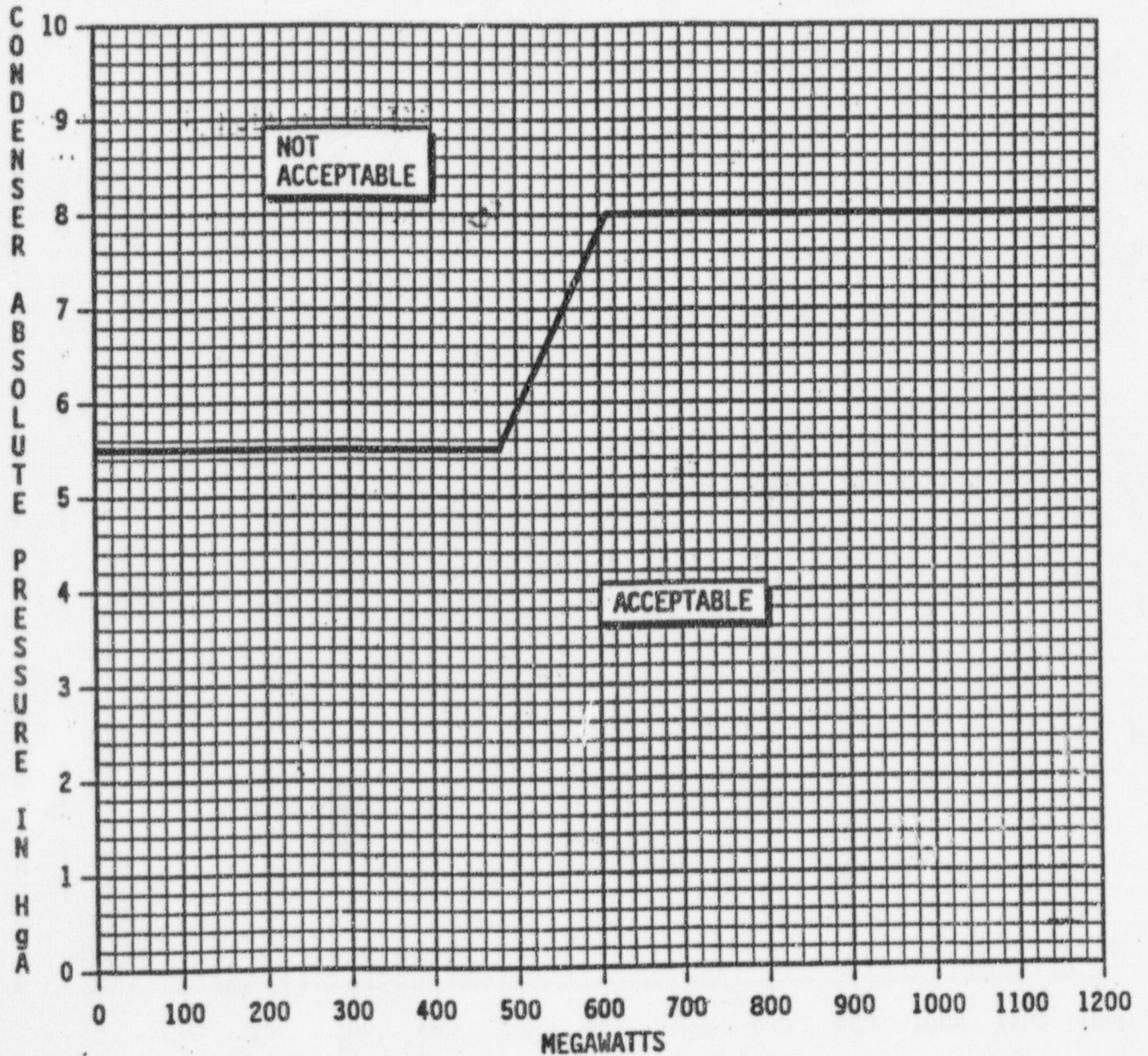


FIGURE 1BwOA SEC-3-1
TURBINE LOAD -vs- CONDENSER PRESSURE

APPROVED

APR 21 1994

BRAIDWOOD
ON-SITE REVIEW

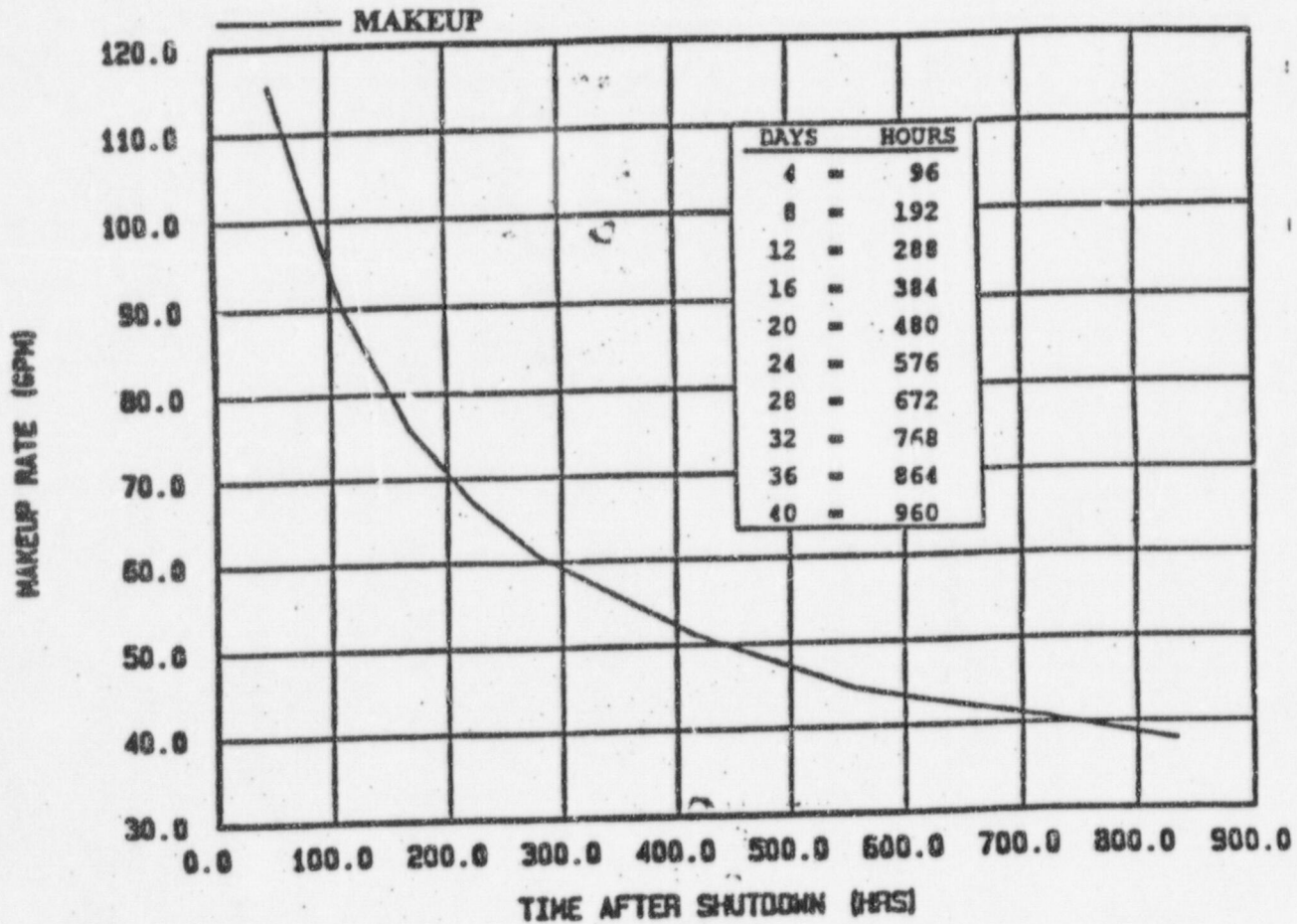


FIGURE 1BwOA PRI 10-3

Minimum Makeup Flow Required to Match Boiloff

STEP

ACTION/EXPECTED RESPONSE

RESPONSE NOT OBTAINED

5 VERIFY ALL CONTROL RODS FULLY
INSERTED:

- All rod bottom lights - LIT

Perform the following:

- a. IF two or more rods are NOT fully inserted, THEN emergency borate 1200 GAL (3600 GAL FROM RWST) for each rod NOT fully inserted per 1BwOA PRI-2, EMERGENCY BORATION.
- *b. Within 1 HOUR calculate Shutdown Margin per 1BwOS 1.1.1.1.e-1, SHUTDOWN MARGIN VERIFICATION DURING SHUTDOWN (1BwOSR 3.1.1.1):

APPROVED
JUN 10 1998

BRAIDWOOD
ON SITE REVIEW