EXAM REPORT W/AS-GIVEN WRITTEN EXAMINATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

October 19, 1999

Charles M. Dugger, Vice President
Operations - Waterford 3
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

SUBJECT: NRC INSPECTION REPORT NO. 50-382/99-14

Dear Mr. Dugger:

This refers to the inspections conducted on September 13-17 and October 4-8, 1999, at the Waterford Steam Electric Station, Unit 3 facility. The inspections evaluated the effectiveness of your licensed operator requalification program and evaluated five applicants for issuance of reactor operator licenses. The enclosed report presents the results of this inspection.

The licensed operator requalification program was effective, utilized a systems approach to training, and showed continued improvements over previous inspection findings. Additionally, all initial license applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

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

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

John L. Pellet, Chief
Operations Branch
Division of Reactor Safety

Docket No.: 50-382
License No.: NPF-38

Enclosure:
NRC Inspection Report No.
50-382/99-14

cc **w/Enclosure and Attachments 1-3**
Jay O'Hern, Training Manager
Waterford Steam Electric Station
P. O. Box B
Killona, LA 70066

cc **w/Enclosure and Attachment 1 only:**
Executive Vice President and
Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Vice President, Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Wise, Carter, Child & Caraway
P.O. Box 651
Jackson, Mississippi 39205

General Manager, Plant Operations
Waterford 3 SES
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

Manager - Licensing Manager
Waterford 3 SES
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

Chairman
Louisiana Public Service Commission
One American Place, Suite 1630
Baton Rouge, Louisiana 70825-1697

Director, Nuclear Safety &
Regulatory Affairs
Waterford 3 SES
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

Ronald Wascom, Administrator
and State Liaison Officer
Louisiana Radiation Protection Division
P.O. Box 82135
Baton Rouge, Louisiana 70884-2135

Parish President
St. Charles Parish
P.O. Box 302
Hahnville, Louisiana 70057

Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

E-Mail report to D. Lange (DJL)
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Regional Administrator	Wat-3 Resident Inspector
DRP Director	RIV File
DRS Director	RITS Coordinator
Branch Chief (DRP/D)	Project Engineer (DRP/D)
Branch Chief (DRP/TSS)	

bcc:**w/Enclosure and Attachments 1-3:**
Laura Hurley
R. Gallo, HOLB/DRCH

DOCUMENT NAME: R:_WAT\WT914RP.REL.WPD *Previously concurred
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/99-14
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	September 13-7 and October 4-8, 1999
Inspector(s):	Ryan Lantz, Lead Inspector/Chief Examiner Gary Johnston, Senior Reactor Inspector/Examiner
Approved By:	John L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	Initial Written Examination
Attachment 3:	Answer Key and Post Administration Comments

EXECUTIVE SUMMARY

Waterford Steam Electric Station, Unit 3
NRC Inspection Report No. 50-382/99-14

This inspection included a review of the licensed operator requalification training program following the guidelines in Inspection Procedure 71001, "Licensed Operator Requalification Program Evaluation." The facility's annual operating test and biennial written examination for licensed operator requalification were administered to one shift crew September 13-15, 1999, during the inspection period. Requalification examinations were also administered the 2 weeks prior to the inspection.

This inspection also evaluated the competency of five reactor operator license applicants for issuance of operating licenses at Waterford, Unit 3. The licensee developed the initial examinations using NUREG-1021, Revision 8. NRC examiners reviewed and approved the examinations. The initial written examinations were administered to all applicants on October 1, 1999, by facility proctors in accordance with the guidance in NUREG-1021, Revision 8. The NRC examiners administered the operating tests on October 4-7, 1999.

Operations

- The licensee developed and submitted a high quality examination, which was administered, with only minor changes. All five applicants passed the initial licensing examinations. No broad knowledge or training weaknesses were identified. With minor exceptions, strengths were observed in communication, peer checks, and procedure usage (Sections O1.1, O4.2, O4.3, and 05.4).

- Good operator performance was observed in all aspects of the requalification examinations, with some exceptions noted. Communications, procedure use, and peer checks were noted strengths. Overall, the operator requalification training program effectively implemented a systems approach to training with several improvements noted. The most significant improvements included stronger operations management involvement in observation and evaluation of licensed operator training, improved effectiveness of training review group meetings, and diversity and availability of program feedback methods (Sections O4.1 and O5.3).

- The licensed operator requalification facility evaluators administered and evaluated the examinations professionally. The inspectors concurred with the facility evaluations (Sections O5.1).

- Requalification examinations were comprehensive, discriminated at the appropriate level, and were responsive to identified weaknesses (Sections O5.2).

- Excellent examination security was maintained (Sections O5.5).

- Licensed operator requalification facility audits were comprehensive, self-critical, and identified areas for potential improvement (Sections O7.1).

- The licensee maintained appropriate operator license conditions by controlling all medical qualification and watch-standing requirements (Sections O7.2).

Report Details

Summary of Plant Status

The facility was shutdown for the duration of the requalification program inspection due to repair efforts on the 2B reactor coolant pump seals. The facility conducted a startup prior to resumption of the initial examinations and was at 100 percent power for the remainder of the inspection. All plant observations were conducted during the day shift, coincident with conduct of the walkthrough examinations.

I. Operations

O1 Conduct of Operations

O1.1 General Observations

a. Inspection Scope

During the in-plant task walkthrough section of the examinations, the examiners observed the on-shift operators during routine operations of the facility.

b. Observations and Findings

The on-shift control room crews performed their duties professionally while supporting administration of the initial examinations in the control room. Communication and annunciator response in the control room was formal and consistent with observations of crews in the simulator during the requalification and initial examinations.

c. Conclusions

The control room staff exhibited good operating practices and were consistent with crews observed during examinations.

O4 Operator Knowledge and Performance

O4.1 Operator Performance on Annual Requalification Examinations

a. Inspection Scope (71001)

The inspectors observed the performance of one shift crew of licensed operators during the dynamic simulator and job performance measure portions of the annual requalification operating examination. The inspectors also reviewed the results of the biennial written examinations.

b. Observations and Findings

The crew in the dynamic simulator consisted of two reactor operators, two senior operators, and one nonlicensed shift engineer who acted as the shift technical adviser. Each licensed operator was administered five job performance measures and participated in two dynamic simulator scenarios. The inspectors observed the licensee staff administer the dynamic simulator scenarios and some job performance measures.

The inspectors observed generally good examinee performance during the job performance measure section of the operating test, including frequent peer review checks and appropriate procedure usage.

During the dynamic simulator portion of the examinations, the inspectors observed consistent three-legged communications, formal procedure usage, and frequent peer checks. Operators in the control room supervisor position displayed good direction and oversight of crew activities. Minor instances were observed where information was not shared with all members of the crew, which did not significantly delay crew response or event mitigation.

The inspectors observed consistent and appropriate use of alarm response procedures, abnormal operating procedures, and the emergency operating procedures.

The biennial written examination was administered by facility proctors on September 16, 1999. One reactor operator and one senior operator failed the written examination. A facility post administration examination review resulted in the deletion of two questions for no correct answer. The deletion of the two questions did not affect the results of the examination grading. The two operators were removed from licensed duties pending successful completion of a remedial written examination to be administered the following week. The two operators passed their remedial written examinations on September 23, 1999, and were returned to licensed duties.

The inspectors noted that in the 3 weeks following the on-site inspection, three other licensed operators failed their examinations: two were written failures, and one failed the system tasks portion of the operating test. These three licensed operators were also removed from licensed duties, remediated and retested, then returned to licensed duties. The inspectors considered the low number of operating test failures and four written exam failures to be consistent with expectations for a challenging requalification examination.

c. Conclusions

Good operator performance was observed in all aspects of the requalification examinations with minor exceptions noted. Communications, procedure use, and peer checks were noted strengths.

04.2 Initial Written Examination

a. Inspection Scope

On October 1, 1999, the facility licensee proctored the administration of the written examinations, as approved by the NRC, to five individuals who had applied for initial reactor operator licenses. The licensee proposed grades for the written examinations and evaluated the results for question validity and generic weaknesses. The examiners reviewed the licensee's results.

b. Observations and Findings

All applicants passed the written examination. The minimum score was 80 percent. After final grading by the chief examiner, the overall average score was 86.3 percent. Applicant scores ranged from 80.6 to 91.8 percent. The licensee's post administration analysis identified that 10 questions were missed by 50 percent or more of the applicants. These were Questions 1, 17, 22, 29, 35, 42, 44, 45, 57, and 80. Review of the 10 questions did not identify any generic training or knowledge weaknesses.

The licensee performed a detailed post administration examination analysis and recommended deletion of four questions from the examination. The analysis and recommendations are attached to this report in Attachment 3. The first question, number 29, was missed by 4 of the applicants and required them to recognize which component listed was powered from the AB DC bus. The facility recommended deletion of the question based on training that was conducted on an abnormal operating procedure for loss of the AB DC bus, noting that the procedure had not been revised to reflect a plant modification that changed the power supply for the charging header isolation valve, CVC-209, to the AB DC bus. Training had been conducted on the power supply change modification several months earlier. The chief examiner agreed to delete the question, not because of the reasons listed by the facility, but because the valve in question was infrequently operated (only for maintenance or if a leak in the charging header developed) and knowledge from memory of the power supply for that valve was considered beyond the recall level of knowledge required for the license.

The second question, number 42, was missed by 3 of the applicants. The facility recommended deletion of the question because the numerical range of the answer choices were all within the reasonable tolerance when reading the provided graph to calculate the correct answer. The chief examiner reviewed the graphs and applied up to a 10 percent reading error, which resulted in two of the answer choices being accepted as correct. All of the applicants chose one of the two answers and, hence, all passed the question.

The third question, number 57, was missed by all of the applicants. The facility recommended deletion of the question based on no correct answer. The chief examiner agreed with the analysis and deleted the question.

The fourth question, number 80, was missed by 3 of the applicants. The facility recommended deletion of the question based on no correct answer. The chief examiner reviewed the question and references provided and retained the question unchanged.

The three other choices in the question were clearly incorrect, and the answer as given was supported by the references as an acceptable action, although not specifically required by the reference.

c. Conclusions

All five applicants passed the initial licensing written examinations. No broad knowledge or training weaknesses were identified.

04.3 Initial Operating Test

a. Inspection Scope

The examination team administered the various portions of the operating examination to the five applicants on October 4-7, 1999. Each applicant participated in two dynamic simulator scenarios. Each applicant received a walkthrough test, which consisted of tasks in 10 systems and tasks or questions in four administrative areas.

b. Observations and Findings

All applicants passed all portions of the operating test. Overall, the applicants performed well in the dynamic simulator scenarios with communications, peer checks, and procedure usage noted as strengths. Similar to observations during the requalification examinations, minor instances were noted where information was not shared with all members of the crew, without significant consequence.

In one instance, the crew failed to evaluate the significance of a failed pressurizer pressure instrument, which resulted in a more rapid depressurization of the reactor coolant system. The crew mistakenly attributed the rate to a leak from the reactor coolant system. Later in the scenario, the reactor operator was directed to change the controlling channel from the failed instrument channel. He neither communicated understanding of why he was taking that action nor did he followup with the senior operator on expected results of the action.

c. Conclusions

All five applicants passed the operating tests. With minor exceptions, the applicants performed well and displayed strengths in communications, peer checks, and procedure usage.

O5 Operator Training and Qualification

05.1 Requalification Examination Administration

a. Inspection Scope (71001)

The inspectors observed the administration of all aspects of the annual operating test to determine the evaluators' ability to administer an examination and assess adequate

performance through measurable criteria. The inspectors conducted interviews to determine the knowledge level of the managers, supervisors, and instructors with regard to the requalification program implementation. The inspectors also observed the fidelity of the plant simulator to support training and examination administration.

b. Observations and Findings

Six training staff evaluators were observed administering the examinations, including preexamination briefings, observations of operator performance, individual and group evaluations of observations, techniques for job performance measure cuing, and final evaluation documentation. The evaluators conducted the examinations professionally with no inadvertent cuing and were thorough in their documentation of observed weaknesses and areas for improvement. The inspectors also noted that the timing of the malfunctions during the dynamic simulator evaluations enabled evaluation of senior operator competencies, such as crew direction and oversight, and ability to prioritize and integrate plant status. The plant simulator fidelity supported the examinations, and no fidelity issues were observed.

The inspectors observed the post scenario crew evaluation process. The evaluations were led by the designated lead evaluator with the operations manager participating in the evaluation. Each of the evaluators participated in discussions of event observations, and strengths and weaknesses observed. The operations manager provided immediate operations feedback for expectations, as well as, an independent observation of crew actions. The inspectors noted that the involvement of an operations department representative in examination observations and evaluations was a significant strength of the training program.

The evaluators passed all of the operators on all portions of the operating tests. The inspectors concurred with the facility evaluations.

c. Conclusions

The licensed operator requalification facility evaluators administered and evaluated the examinations professionally. The inspectors concurred with the facility evaluations.

05.2 Requalification Biennial Written Examination and Annual Operating Tests

05.2.1 Requalification Written Examinations

a. Inspection Scope (71001)

The inspectors reviewed the biennial comprehensive written examination that was administered September 15, 1999. The examination was reviewed to evaluate its validity and compliance with facility program requirements. The examinations were reviewed using guidance of NUREG-1021, "Operator Licensing Examiner Standards," for construction, scope of coverage, and cognitive level.

b. Observations and Findings

The biennial written examination consisted of 40 multiple choice questions, half of which focused on system knowledge. The other half generally focused on administrative controls and Technical Specification applications. The examination questions tested at the appropriate level of comprehension were linked to important learning objectives and followed the established sample plan. The sample plan was representative of the training accomplished over the training cycle and specifically identified senior operator only questions. The inspectors reviewed the designated questions and agreed that they were senior operator level questions, sampled appropriately from 10 CFR 55.43 items, and met the program requirements as described in NTI-OP-004, "Examination Development."

c. Conclusions

The comprehensive biennial written requalification examinations adequately sampled from accomplished training and were appropriate in scope, depth, and cognitive level.

05.2.2 Requalification Annual Operating Test

a. Inspection Scope (71001)

The inspectors reviewed the annual operating test administered on September 14 and 15, 1999, as well as the examinations given during the first 2 weeks of requalification examinations. The inspectors used the guidance in NUREG-1021 to determine the validity of the examinations to provide a basis for evaluating the examinee's knowledge of abnormal and emergency operating procedures and operation of plant systems.

b. Observations and Findings

The inspectors noted that system and administrative tasks and simulator scenarios were adequate for evaluation of operator and crew skills. The system and administrative tasks contained clear, measurable performance standards and critical task acceptance criteria. The scenarios contained clearly stated objectives and crew critical tasks. Scenario initial conditions were realistic and events were a logical progression of related events. Guidelines of NUREG-1021 for qualitative attributes of job performance measures and dynamic simulator scenarios were met, although the degree of challenge and level of difficulty varied notably between operating tests given in different weeks. The inspectors did not consider this a programmatic vulnerability, only a consistency issue with quality reviews of the examinations.

c. Conclusions

The annual operating examinations were comprehensive and discriminated at the appropriate level.

O5.3 Review of Requalification Program Guidance

a. Inspection Scope (71001)

The inspection evaluated the effectiveness of the methodology used to develop and construct the requalification examinations, assessed the effectiveness of the examinations to identify retraining needs, and measure the examinees' subject knowledge. The inspectors utilized the guidance of NUREG-1220, "Training Review Criteria and Procedures," Revision 1, and 10 CFR 55.59, "Requalification," to evaluate the requalification program's ability to meet Elements 4 and 5 of a systems approach to training. This evaluation included review of the feedback tracking system and remedial training program and interviews with training management, instructors, evaluators, and examinees.

b. Observations and Findings

The licensee's training staff indicated that the guidelines of NUREG-1021 were substantially utilized for the development and administration of the licensed operator requalification examination, in addition to the facility requalification training procedures. The licensee supplemented the guidance of NUREG-1021 with training instructions for the development, control, and administration of licensed operator requalification examinations and quizzes, 2-year training plan development, makeup training, and remediation guidelines. The licensee's procedures were comprehensive.

The inspectors noted that program guidance in NTI-OP-008, "Licensed Operator Requalification," contained specific guidance for development and security of examinations, post-examination activities, removal from shift for failures of examinations, and remedial training of individuals. The remedial programs focused on identified weaknesses and adequately retested the operators prior to returning them to licensed duties.

The inspectors noted that program revisions had been made to address recent audit and self-assessment findings. The most significant of these changes involved mandatory operations management participation in evaluated dynamic simulator crew examinations and quizzes, a modification of the instructor's feedback mechanism, and a more defined requirement for training review group meeting attendance and frequency.

c. Conclusions

The licensed operator requalification program effectively implemented a systems approach to training to identify operator performance and training weaknesses and modify the training and evaluation program to correct identified weaknesses.

05.4 Initial Licensing Examination Development

The facility licensee developed the initial licensing examination in accordance with guidance provided in NUREG-1021, "Operating Licensing Examination Standards," Revision 8.

05.4.1 Examination Outline

a. Inspection Scope

The facility licensee submitted the initial examination outlines on June 29, 1999. The chief examiner reviewed the submittal against the requirements of NUREG-1021, Revision 8.

b. Observations and Findings

The initial examination outlines met the qualitative and quantitative guidelines of NUREG-1021. The chief examiner provided minor enhancement comments and advised the licensee to proceed with examination development.

c. Conclusions

The licensee submitted examination outlines were adequate as presented with minor enhancement comments provided by the chief examiner.

05.4.2 Examination Package

a. Inspection Scope

The draft examinations were transmitted by the licensee to the NRC on August 31, 1999. The licensee submitted the completed final examination package on September 28, 1999, following the chief examiner's on-site review. The chief examiner reviewed the examinations against the requirements of NUREG-1021, Revision 8.

b. Observations and Findings

The licensee submitted a technically valid draft examination of excellent quality that was responsive to the outline. Following two independent examiner reviews, the chief examiner provided enhancement suggestions for 10 questions and other editorial suggestions. The enhancement comments generally related to distractor plausibility or adding detail to the question stem. After discussion of the examiner review comments, the licensee modified the examinations as agreed.

The licensee submitted four scenarios including a backup that were of excellent quality. The NRC reviewers had no significant comments following in-office reviews. The examiners validated the scenarios during the week of September 13, 1999, and determined that one of the scenarios required an additional instrument failure with substantial action by the reactor operators. Otherwise, the scenarios were adequate as proposed and fully supported by the simulation facility.

To support the systems walkthrough section of the operating test, the facility licensee provided 10 job performance measures (JPMs) of excellent quality to evaluate selected operator tasks. The chief examiner required replacement of one task based on its focus on administrative rather than system knowledge. The examiners provided additional

minor enhancement comments to the remainder of the tasks during the examiner's onsite review.

To support the administrative topics section of the operating test, the licensee submitted a mix of administrative questions and tasks. The examiners provided minor editorial comments and one enhancement comment to correct a technical error in one task during the chief examiner's on-site review.

c. Conclusions

The licensee developed and submitted a high quality examination, which was administered with only minor changes.

05.5 Examination Security

a. Scope

The examiners reviewed examination security during the examination administration week for compliance with NUREG-1021 requirements.

b. Observations and Findings

Members of the licensee's operations and training staff signed onto the NUREG-1021 examination security agreement, acknowledging their responsibilities for examination security. Daily security briefings were conducted for the examinees. Examination security plans were effective and properly implemented. Applicants were maintained under a constant escort as needed to prevent any actual or appearance of compromise. Simulator security was strictly maintained.

c. Conclusions

Excellent examination security was maintained.

O7 Quality Assurance in Operator Training

O7.1 Review of Operations Quality Assurance Audits

a. Inspection Scope (71001)

The inspectors reviewed the results of "Waterford-3 Training Programs Combined Assessment/QA Audit," May 11-15, 1998, which included an audit of the licensed operator requalification program. The inspectors reviewed additional audits of the licensed operator requalification program dated June 14 and July 12-15, 1999, that were conducted by peer groups and outside organizations.

b. Observations and Findings

The quality assurance audit and the additional audits both utilized personnel from outside the licensee's organization. All three audits were comprehensive and critical with items for improvement noted in the reports. Training requests were initiated and tracked regarding the identified areas for improvement.

c. Conclusion

The licensed operator requalification program audits were comprehensive, self-critical, and identified areas for potential improvement.

07.2 Maintenance of Licensed Operator Qualifications

a. Inspection Scope (71001)

The inspection determined the effectiveness of the licensee's program to monitor and control licensed operator proficiency watches and medical qualifications. The inspectors utilized the guidance of ANSI/ANS 1.134, "Medical Evaluation of Licensed Personnel for Nuclear Power Plants, Revision 2; 10 CFR 55.21, "Medical Examination"; and 10 CFR 55.23, "Certification," with regard to medical qualifications. 10 CFR 55.53, "Conditions of Licenses," provides guidance on requirements for maintaining an active license and the actions required to reactivate when a licensed operator is in the inactive status.

b. Observations and Findings

Licensed operators were required by OI-024-000, "Maintaining Active SRO/RO Status," to perform five proficiency watches each quarter to maintain an active license. A review of the licensee computer-based tracking system indicated that all active licensed operators were completing at least five watches per quarter. The inspectors also determined that licensed operators were performing those watches for a full 12-hour period and had stood those watches in the appropriate licensed position for which the individual was licensed. Licensee management indicated that it is the individual licensed operators' responsibility to schedule and maintain proficiency watch status. However, operations management monitors the watch standing closely through periodic checks of the status in the computer-based tracking system.

Similarly, a computer tracking data base is maintained for training and medical qualifications. The inspector reviewed the current status of licensed operator qualifications and determined that all currently licensed operators have maintained their qualifications. A review of eight individual medical record files found that the medical examinations were conducted annually and conformed with guidance in ANSI/ANS 1.134, "Medical Evaluation of Licensed Personnel for Nuclear Power Plants." The inspector noted that the licensee emphasized in a recent licensed operator newsletter, operators' responsibilities with regard to keeping medical and operations management informed of any changes in their medical status. This was apparent with several operators having recent changes to conditions in their licenses due to various surgical corrections to their

eyes. Each change was promptly reported to the facility medical staff, and the change in condition was also reported to the NRC via NRC form 396.

c. **Conclusions**

The licensed operators maintained appropriate license conditions by meeting all medical qualification and watch-standing requirements.

IV. Plant Support

F8 Miscellaneous Fire Protection Issues

F8.1 General Comments

The inspectors observed general plant housekeeping incident to administration of the in plant job performance section of the operating test. The facility was reasonably clean and well lighted and the floors were clear and free from debris. The applicants were conscientious to note discrepancies and inform the main control room.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management and staff at the conclusion of the inspection on October 8, 1999. An interim exit meeting for the requalification program inspection was held on September 16, 1999. The licensee acknowledged the findings presented.

The licensee did not identify as proprietary any information or materials examined during the inspection.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. DeDeaux, Licensing Supervisor
E. Ewing, Director, Nuclear Safety Assurance
R. Fili, Quality Assurance Manager
R. Fletcher, Operations Training Supervisor
J. Hoffpauir, Operations Manager
T. Leonard, General Manager, Plant Operations
D Madere, Site Support
J. O'Hern, Director, Training
O. Pipkins, Licensing Engineering

NRC

P. Gage, Senior Reactor Inspector, RIV

LIST OF INSPECTION PROCEDURES USED

IP 71001 Licensed Operator Requalification Program Evaluation

PROCEDURES REVIEWED

Training Audits

- "Waterford 3 Training Assessment," July 12-15, 1999
- "WESTRAIN Assist for NRC 71001 Audit," June 14, 1999.
- "Waterford-3 Training Programs Combined Assessment/QA Audit," May 11-15, 1998.

Training Procedures

- NTP-101, "Operations Training Programs," Revision 10, June 24, 1999.
- NTP-006, "Control and Administration of Examinations," Revision 9, May 27, 1999.
- NTI-OP-004, "Examination Development," Revision 8, September 12, 1999.
- NTI-OP-005, "Examination Administration," Revision 6, January 6, 1998.
- NTI-OP-008, "Licensed Operator Requalification," Revision 4, August 23, 1999.

Operating Procedures

- OP-001-003, "Reactor Coolant System Draindown," Revision 18
- OP-901-313, "Loss of a 125V DC Bus," Revision 1
- OI-024-000, "Maintaining Active SRO/RO Status," Revision 5

Other Documentation

- Condition Report 1999-0939, September 14, 1999
- Training Review Group Meeting Minutes of May 20, July 8, and August 19, 1999.
- 1999 Biennial Written Examinations W-3-EXM-LOR-9951-3
- 1999 Annual Operating Tests, Weeks 1-3.

ATTACHMENT 2

INITIAL WRITTEN EXAMINATION

Waterford 3 Examination Question
Examination Bank

Examination Question Number 1

Description:	Downpower with Turbine and boration not matched			**Question ID:**	5656
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-010-005	0	0	2/5/99	CVC, RCS

NRC KA Number	RO	SRO	Training Material	Objective
3.2-002-A1.03	3.7	3.8	W-3-LP-OPS-PPN02	3

QUESTION

A plant downpower is in progress. T_{REF} if 565°F and dropping. T_{AVE} is 570°F and steady. Direct boration is in progress at 20 gpm, and the Main Turbine is unloading at 30 MW/min. Select which is true regarding this downpower.

A. Turbine load rate is excessive for the boration rate, as evidenced by the steady T_{AVE} indication.

B. Turbine load rate is excessive for the boration rate, as evidenced by dropping T_{REF} indication.

C. Boration rate is excessive for the turbine load rate, as evidenced by the dropping T_{REF} indication.

D. Boration rate is excessive for the turbine load rate, as evidenced by the steady T_{AVE} indication.

ANSWER

A Turbine load rate is excessive for the boration rate, as evidenced by the steady T_{AVE} indication.

COMMENTS

Examination Question Number 2

Description:	Raising SIT level to raise boron concentration			**Question ID:**	5657
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-009-008	15	0	7/7/99	SI

NRC KA Number	RO	SRO	Training Material	Objective
3.2-006-A1.13	3.5	3.7	W-3-LP-OPS-SI00	5

QUESTION

The results of a routine boron sample of Safety Injection Tank 2B are out of spec low. The CRS directs you to drain and refill SIT 2B from the RWSP. You are required to do **ALL** of the following **EXCEPT**:

A. Contact the System Engineer to calculate an estimated boron concentration.

B. Allow the HPSI return header to recirculate back to RWSP for 10 minutes.

C. Document in the Station Log that the HPSI header sample results are in specification as determined by Chemistry Department.

D. Recirculate the RWSP for a minimum of 5 hours with the LPSI Pump or 20 hours with the HPSI Pump prior to filling the required SIT.

ANSWER

D Recirculate the RWSP for a minimum of 5 hours with the LPSI Pump or 20 hours with the HPSI Pump prior to filling the required SIT.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 3

Description:	HPSI injection on loss of SDC.			**Question ID:**	5679
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-131	1	1	2/9/99	RCS, SI

NRC KA Number	RO	SRO	Training Material	Objective
3.2-006-K1.03	4.2	4.3	W-3-LP-OPS-SI00	2, 6

QUESTION

A loss of Shutdown Cooling event is in progress. No LPSI Pumps are running. HPSI Pump A was started IAW OP-901-131, Shutdown Cooling Malfunction for RCS makeup. Which of the following is correct concerning restoring a Shutdown Cooling Train?

A. Vent and start LPSI Pump A, since HPSI Pump A is injecting to Hot Leg 1.

B. Vent and start LPSI Pump A, since HPSI Pump A is injecting to Hot Leg 2.

C. Vent and start LPSI Pump B, since HPSI Pump A is injecting to Hot Leg 1.

D. Vent and start LPSI Pump B, since HPSI Pump A is injecting to Hot Leg 2.

ANSWER

C. Vent and start LPSI Pump B, since HPSI Pump A is injecting to Hot Leg 1.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 4

Description:	Pressurizer Pressure Control			**Question ID:**	5658
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
Sys Desc for PLC				PPC

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.3-010-A3.02	3.6	3.5	W-3-LP-OPS-PLC00	3, 5

QUESTION

Pressurizer level is recovering following a Letdown malfunction. As level rises, the Pressurizer Pressure Control System responds by:

A. Pressurizer Pressure Controller output rises, Pressurizer Spray Controller output rises, Pressurizer Proportional Heaters' current 0 amps, and the Spray Valves throttle to maintain pressure.

B. Pressurizer Pressure Controller output rises, Pressurizer Spray Controller output rises, Pressurizer Proportional Heaters' current at maximum, and the Pressurizer Back-up Heaters cycle on to maintain pressure.

C. Pressurizer Pressure Controller output drops, Pressurizer Spray Controller output drops, Pressurizer Proportional Heaters' current 0 amps, and the Spray Valves throttle to maintain pressure.

D. Pressurizer Pressure Controller output drops, Pressurizer Spray Controller output drops, Pressurizer Proportional Heaters' current at maximum, and the Pressurizer Back-up Heaters cycle on to maintain pressure.

ANSWER

A

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 5

Description:	PZR Level Control Malfunction			**Question ID:**	1484
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	Used Bank	**History:**	RO class 4-16-99		

Reference	Revision	Change	Date	Plant Systems:
OP-901-110	3	1	9/2/98	PLC, PPO

NRC KA Number	RO	SRO	Training Material	Objective
3.2-011-K4.01	3.3	3.7	W-3-LP-OPS-PLC00	06
3.2-011-K6.03	2.9	3.3		
3.2-011-K1.01	3.6	3.9		

QUESTION

- The plant is operating at full power.
- Pressurizer level control is selected to channel X.
- The low level heater cutoff switch is selected to BOTH.
- Channel X level indicates 0%
- Channel Y level indicates 55%.

Which of the following automatic actions will occur?

A. Backup charging pump stops if running, letdown flow goes to minimum, pressurizer heaters trip.

B. Backup charging pumps start, letdown flow increases, pressurizer heaters full on.

C. Backup charging pump stops if running, letdown flow increases, pressurizer proportional heaters full on.

D. Backup charging pumps start, letdown flow goes to minimum, pressurizer heaters trip

ANSWER

D. Backup charging pumps start, letdown flow goes to minimum, pressurizer heaters trip

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 6

Description:	Automatic bypassing of the high log power trip			**Question ID:**	5678
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-903-102	09	05	3/30/99	PPS

NRC KA Number	RO	SRO	Training Material	Objective
3.7-012-K4 06	3.2	3.5	W-3-LP-OPS-PPS00	4

QUESTION

The high log power trip bypass is automatically removed on any channel that power drops below:

A. $1x10^{-6}$, as detected by the safety log power for that channel

B. $8.5x10^{-5}$, as detected by PAC for that channel

C. $1x10^{-4}$, as detected by the safety log power for that channel

D. $5x10^{-4}$, as detected by the CPC for that channel.

ANSWER

C. $1x10^{-4}$, as detected by the safety log channel

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 7

Description:	CWP for Target CEA's vs. CEAC's			**Question ID:**	3081
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	Used Bank	**History:**	RO class 9/98		

Reference	Revision	Change	Date	Plant Systems:
OP-004-003	06	02	10/13/98	CPC

NRC KA Number	RO	SRO	Training Material	Objective
3.1-001-K4.08	3.2*	3.4	W-3-LP-OPS-CPCO1	4
3.1-014-K4.06	3.4	3.7		

QUESTION

You are withdrawing Reg Group 6 CEAs, IF one CEA in the group stops moving while the rest continue to withdraw, WHICH of the following will prevent any further group withdrawal when reached (assume Group 6 is at 130 inches when misalignment occurs)

A. 4.95 inch misalignment detected by CPC target CEAs

B. 5.5 inch misalignment detected by CEACs

C. 5.5 inch misalignment detected by CPC target CEAs

D. 4.95 inch misalignment detected by CEACs

ANSWER

B. 5.5 inch misalignment detected by CEACs

COMMENTS

Examination Question Number 8

Description:	Tcold fails high, Pressurizer level setpoint fails, CVCS responds			**Question ID:**	5659
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-110	3	1	9/2/98	CVC

NRC KA Number	RO	SRO	Training Material	Objective
3.7-016-K3.02	3.4*	3.5*	W-3-LP-OPS-PLC00	7

QUESTION

A reactor start-up is in progress. The reactor is not yet critical. Reactor Reg 2 is selected on CP-2 and the T_{COLD} selector switches for loop 1 and 2 are in NORM. The T_{AVE} LOOP SELECTOR on both Reactor Reg drawers 1 and 2 are in BOTH. What is the proper plant response if the normal T_{COLD} RTD for loop 1 fails high?

A. No change due to Reactor Reg 2 being selected.

B. Letdown flow goes to maximum.

C. Letdown flow goes to minimum, both back-up Charging Pumps start.

D. No change due to setpoint failing to current Pressurizer level.

ANSWER

C

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 9

Description:	Actions on RAS to minimize off site and in-plant exposure			**Question ID:**	5660
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-002	08	0	9/9/98	SI

NRC KA Number	RO	SRO	Training Material	Objective
3.5-026-K4.08	4.1*	4.3*	W-3-LP-OPS-PPE02	11, 25

QUESTION

A large break LOCA has occurred, and a RAS has actuated. What actions must be taken to minimize radiation exposure to plant and off site personnel?

A. Verify LPSI Pumps A and B are secured after verifying adequate Safety Injection Sump level.

B. Secure all Charging Pumps and isolate the Charging penetration.

C. Close SI-106 A and B, ESF Pumps' suction from the RWSP.

D. Close SI-120 A (B) and SI-121 A (B), SI Recirculation isolations to the RWSP.

ANSWER

D

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 10

Description:	Containment Spray riser level and Tech Spec compliance			**Question ID:**	5680
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-903-001	22	3	7/26/99	CS
TS 3.6.2				

NRC KA Number	RO	SRO	Training Material	Objective
2-1-33	3.4	4	W-3-LP-OPS-CS00	6

QUESTION

The plant is operating in mode 4 with the following conditions:

- RCS pressure is 410 psia
- Containment Spray Train B is aligned for mode 1-4 operation
- Containment Spray Train B riser level is 185 ft
- Barometric pressure, from the PMC, is 30.01 inhg
- Average Containment pressure is 15.07 psia

Which of the following is correct regarding Containment Spray Train B.

A. Containment Spray Train B is INOPERABLE due to riser level below tech spec minimum level.

B. Containment Spray Train B is not required for operability with RCS pressure >400 psia.

C. Containment Spray Train B is OPERABLE.

D. Containment Spray Train B is INOPERABLE due to RCS pressure being >400 psia in mode 4.

ANSWER

C. Containment Spray Train B is OPERABLE

COMMENTS

Supply student with OP-903-001, attachment 11.1 and 11.17 and Tech Spec 3.6.2.1

Waterford 3 Examination Question
Examination Bank

Examination Question Number 11

Description:	Which signals automatically isolate Containment Purge.			**Question ID:**	5662
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-002-010	13	2	9/18/98	HVR

NRC KA Number	RO	SRO	Training Material	Objective
3.8-029-k1.03	3.6	3.8	W-3-LP-OPS-HVR00	1, 4

QUESTION

Containment Purge will be automatically secured on all of the following **EXCEPT**:

A. HI-HI alarm signal on either Plant Stack PIG gas channel

B. HI-HI alarm signal on the Containment PIG gas channel

C. Containment Isolation Actuation Signal

D. HI-HI alarm signal on any Containment Purge Area Rad Monitor.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 12

Description:	CC-620 closures			**Question ID:**	5655
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
Sys Desc for CC				CC, FS

NRC KA Number	RO	SRO	Training Material	Objective
3.8-033-A3.01	2.5*	2.7*	W-3-LP-OPS-CC00	3

QUESTION

All of the following will close CC-620, Fuel Pool Heat Exchangers temperature control valve, **EXCEPT**: (consider each item separately)

A. Operator closes CC-200 A and CC-727 with control switch on CP-8

B. Safety Injection Actuation Signal

C. Loss of control power

D. CCW Surge Tank level switch, CC ILS7013 B fails low, closing CC-200 B and CC-563.

ANSWER

C. Loss of control power

COMMENTS

Examination Question Number 13

Description:	Escalating Steam Generator tube leak.			**Question ID:**	5664
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-202	2	1	6/23/99	RCS

NRC KA Number	RO	SRO	Training Material	Objective
3.4-035-A2.01	4.5	4.6	W-3-LP-OPS-PPO20	3

QUESTION

Plant conditions are as follows:

- Main Steam Line 1 Rad Monitor and Condenser Off Gas Rad Monitor readings are elevated and rising.
- At 0930, RCS leakage was determined to be 0.5 gpm from the calculated Steam Generator leakage displayed on PMC Group MCES, and verified by Chemistry samples.
- The CRS has implemented OP-901-202, Steam Generator Tube Leakage or High Activity and ordered a plant shutdown IAW OP-010-005, Plant Shutdown.
- At 1015 the SNPO announces that the leak rate into Steam Generator 1 has risen to 0.75 gpm, as on PMC Group MCES.

Given these conditions, which statement is true?

A. A rapid plant shutdown is required because total leakage into Steam Generator 1 rose by 0.25 gpm in less than 60 minutes.

B. Continue the normal plant shutdown unless leakage into Steam Generator 1 exceeds 1.0 gpm.

C. Continue the normal plant shutdown unless secondary coolant specific activity exceeds 0.1 μci/gm dose equivalent I-131.

D. Trip the Reactor and initiate Safety Injection Actuation and Containment Isolation Actuation.

ANSWER

A

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 14

Description:	Atmospheric Dump Valve failure and plant response			**Question ID:**	5740
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-500-011	10	2	11/11/97	MS

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.4-039-K5.08	3.6	3.6	W-3-LP-OPS-MS00	1, 4

QUESTION

The plant is operating at 100% when the PNPO notices T_{COLD} dropping and Power rising. Which of the following would give these indications?

A. Atmospheric Dump Valve #1 failed open.

B. Emergency Feedwater Pump AB tripped on overspeed.

C. EH-118, EH Emergency Trip Header Interface, fails open.

D. MOOG valve for Governor Valve #4 failed, closing Governor #4.

ANSWER

A. Atmospheric Dump Valve #1 failed open

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 15

Description:	Actions to stabilize the RCS during an ESD			**Question ID:**	5715
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-009	00	00	9/9/98	EFW, PPE, MS

NRC KA Number	RO	SRO	Training Material	Objective
4.4-E5-EA1.2	3.5	3.9	W-3-LP-OPS-PPE04	4

QUESTION

A Main Steam line break has occurred inside Containment on Steam Generator 2. Standard Post Trip actions are in progress. Which one of the following is correct regarding stabilizing the Reactor Coolant System?

A. When standard Post Trip actions are complete, open #1 ADV 100% and manually feed with EFW after EFAS auto actuation.
B. When standard Post Trip actions are complete, open #1 ADV 50% and manually initiate EFAS and begin feeding in manual.
C. When Pressurizer pressure or T_{CET} begin to rise, open #1 ADV 100% and manually initiate EFAS and begin feeding in manual.
D. When Pressurizer pressure or T_{CET} begin to rise, open #1 ADV 50% and manually feed with EFW after EFAS auto actuation.

ANSWER

C. When Pressurizer pressure or T_{CET} begin to rise, open #1 ADV 100% and manually initiate EFAS and begin feeding in manual.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 16

Description:	Loss of 125 volt DC bus, A-DC			**Question ID:**	5665
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-313	1	0	4/19/94	CED, DC

NRC KA Number	RO	SRO	Training Material	Objective
3.6-063-K3.02	3.5	3.7	W-3-LP-OPS-CED00	3,4
			W-3-LP-OPS-PPO30	3

QUESTION

Which of the following sets of Reactor Trip Breakers open as a direct result of de-energizing the 125 volt DC bus, A-DC?

A. 1, 2, 5, 6

B. 3, 4, 7, 8

C. 1, 3, 5, 7

D. 2, 4, 6, 8

ANSWER

C. 1, 3, 5, 7

COMMENTS

Examination Question Number 17

Description:	Emergency Diesel Fuel Oil Storage Tank level drop			**Question ID:**	5667
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
TS 3.8.1				EDG
OP-009-002	17	04	7/8/99	

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.6-064-K6.08	3.2	3.3	W-3-LP-OPS-EDG00	7

QUESTION

EGF-104 A, Diesel Fuel Oil Storage Tank A Drain, has been leaking by its packing. Annunciator "EDG A Fuel Oil Strg Tk Lvl Hi/Lo" alarms and the Outside watch reports that level is reading 94.0%. He has tightened down on the packing, and the leak has stopped. What is the impact on the Emergency Diesel Generator A?

A. Emergency Diesel Generator A is inoperable. Perform OP-903-066 within 1 hour.

B. Emergency Diesel Generator A is still operable for 5 days, provided fuel oil is on site within 48 hours.

C. Emergency Diesel Generator A is operable with no other actions required.

D. Emergency Diesel Generator A is still operable, provided fuel oil is ordered within 48 hours and on site within 5 days.

ANSWER

B. Emergency Diesel Generator A is still operable for 5 days, provided fuel oil is on site within 48 hours.

COMMENTS

Provide student with the Tank Curves Data Book and Tech Spec 3.8.1.1.

Waterford 3 Examination Question
Examination Bank

Examination Question Number 18

Description:	LWM Discharge indications			**Question ID:**	5669
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-007-004	14	03	6/12/98	LWM

NRC KA Number	RO	SRO	Training Material	Objective
3.7-073-A4.01	3.9	3.9	W-3-LP-OPS-LWM00	4

QUESTION

A discharge of Waste Condensate Tank B has been established. Flow is at 47 gpm with the M/A station LWM-IFIC-0647 in automatic. LWM-441 is open and LWM-442 is throttled. Later, the indications at CP-4 show that flow is 0 gpm, LWM-IFIC-0647 output is 100%, and both LWM-441 and 442 are full open. Which best explains what has happened.

A. The LWM Rad Monitor, PRMIRE-0647, reached it's high alarm setpoint and secured the release.

B. Discharge flow reached its high flow setpoint and secured the release.

C. M/A station LWM-IFIC-0647 failed, securing the release.

D. Waste Condensate Pump B tripped on low Waste Condensate Tank level, securing the release.

ANSWER

D Waste Condensate Pump B tripped on low Waste Condensate Tank level, securing the release.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 19

Description:	Actions on loss of Circ Water and dropping vacuum.			**Question ID:**	5671
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-007-004	14	3	6/12/98	CW, LWM

NRC KA Number	RO	SRO	Training Material	Objective
3.8-075-A2.03	2.5	2.7	W-3-LP-OPS-LWM00	8

QUESTION

The plant is in Mode 3 with RCS temperature being controlled by MS-319 A, Main Steam Bypass 1A. Steam Generators are being fed with the Auxiliary Feedwater Pump. Waste Condensate Tank B is being discharged to Circ Water. Circ Water Pumps A & B are running. The SUT A feeder to the A1 bus trips open. Which of the following is correct?

A. MS-319 A will shut if vacuum drops to 14 inhg.

B. The operator is required to immediately open both Atmospheric Dump Valves to 50% open.

C. A complete loss of Main Feedwater will occur if vacuum drops to 14 inhg.

D. The operator is required to immediately secure the Waste Condensate Tank release.

ANSWER

D. The operator is required to immediately secure the Waste Condensate Tank release.

COMMENTS

Examination Question Number 20

Description:	Actuation of ARR fire supression			**Question ID:**	5675
Tier:	2	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-009-004	11	2	5/18/98	FPD, FPP

NRC KA Number	RO	SRO	Training Material	Objective
3.8-086-A4.02	3.5	3.5	W-3-LP-OPS-FP00	2, 10

QUESTION

FP-601 A is closed due to a small leak in the fire header in Containment. Fire alarms are received on the FDMCP for the E-13 A for the Airborne Radiation Removal Unit A. The appropriate actions are:

A. Dispatch the RCA watch to the –4 wing area to actuate FPM-1 and verify FP-601 A automatically opens.

B. Pull the manual pull station on the Control Room north wall to actuate FPM-1 and verify FP-601 A automatically opens.

C. Pull the manual pull station on the Control Room north wall to actuate FPM-1 and open FP-601 A with the control switch on CP-8.

D. Verify that FPM-1 automatically actuated and open FP-601 A with the control switch on CP-8.

ANSWER

C. Pull the manual pull station on the Control Room north wall to actuate FPM-1 and open FP-601 A with the control switch on CP-8.

COMMENTS

Examination Question Number 21

Description:	CEA T.S. applied to Continuous Rod Withdrawal			**Question ID:**	5661
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
TS 3.1.3				CED, TS

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A1-AK3.02	3.2	4.3	W-3-LP-OPS-CED0	8
			W-3-LP-OPS-CED00	9

QUESTION

Reactor power is 100% and ASI control is in progress. The PNPO inserts Reg Group 6. When the PNPO releases the IN-HOLD-OUT switch to the Hold position, CEA 23 continues to insert into the core. The PNPO takes the CEDMCS Mode Selector Switch to OFF and CEA 23 insertion stops. Final position of CEA 23 is 118.5". All other CEAs in Reg Group 6 are at 125.25". Reg Group 6 is placed on the HOLD Bus to troubleshoot the cause of the malfunction. Which of the following actions is correct?

A. Declare CEA 23 inoperable and verify SDM within one hour.

B. Declare CEA 23 inoperable and comply with actions for exceeding Long Term Steady State Insertion Limits.

C. Declare Reg Group 6 inoperable and be in Hot Standby within 6 hours.

D. Declare Reg Group 6 inoperable and reduce power to < 70% within one hour in accordance with the COLR.

ANSWER

B

COMMENTS

Supply T.S. 3.1.3.1-3.1.3.7, including COLR pages, to examinees

Waterford 3 Examination Question
Examination Bank

Examination Question Number 22

Description:	Indications of a Dropped CEA		
Tier:	1	**Group:**	2
Source:	New	**History:**	N/A

Question ID: 5663
Cognitive Level: Comp/Analysis

Reference	Revision	Change	Date
OP-901-102	3	0	12/10/98

Plant Systems:
CED, PPO

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A3-AA2.03	3.6	3.8	W-3-LP-OPS-PPO10	1

QUESTION

All of the following indications would be present if CEA 1 dropped to the bottom of the core at 12.5 EFPD and 100% power, with the exception of:

A. A large Azimuthal Power Tilt

B. A large decrease in RCS temperature

C. A decrease in Turbine first stage pressure

D. A decrease in RCS pressure

ANSWER

A

COMMENTS

Examination Question Number 23

Description:	Verification of Two Phase Natural Circulation Flow			**Question ID:**	5668
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
OP-902-002	8	0	9/9/98

Plant Systems:
PPE, RCS

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E11-EA2.09	4.2	4.3	W-3-LP-OPS-PPE02	11

QUESTION

A large break LOCA occurred 20 minutes ago and the following conditions exist:

- RCS Pressure = 500 psia
- Representative CET Temperature = 600°F
- RVLMS is fluctuating between 20% and 0% plenum level
- EFAS-1 and EFAS-2 are initiated, S/G 1 and S/G 2 levels are between 68% and 71% with EFW flow controllers in automatic
- Individual HPSI flow = 200 gpm/cold leg
- LPSI train A and B flow rate = 0 gpm
- Charging Pump A and B are running, Charging Pump AB is secured with AB assignment switch in NORM

Which of the following is correct concerning the criteria for verifying satisfactory two-phase natural circulation?

A. Representative CET temperature is not met.

B. HPSI flow is not met.

C. All criteria are met.

D. Charging pump status is not met.

ANSWER

A

COMMENTS

Supply Steam Tables and Attachment 2-B of OP-902-009

Examination Question Number 24

Description:	Minimum Pzr level requirements during a charging malfunction			**Question ID:**	5670
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-112	2	0	6/30/97	CVC, RCS, PPO

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A22-AK1.03	3	3.4	W-3-LP-OPS-PPO10	3

QUESTION

The plant is at 100% power with T_{cold} at 545°F and T_{hot} 603°F. The running charging pump tripped on low suction pressure due to the VCT Outlet Valve, CVC-183 closing with no apparent cause. Letdown isolated on high temperature due to the loss of the running charging pump. Determine the minimum required Pressurizer level to remain at power for the given conditions.

A. 6.0%

B. 33.1%

C. 43.0%

D. 55.6%

ANSWER

C

COMMENTS

Supply examinee with copy of Attachment 1 of OP-901-112

Waterford 3 Examination Question
Examination Bank

Examination Question Number 25

Description:	Time to exceed 200 degrees after a loss of Shutdown cooling			**Question ID:**	5672
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
OP-901-131	1	1	2/9/99

Plant Systems:
PPO, SDC

NRC KA Number	RO	SRO	Training Material	Objective
2-4-9	3.3	3.9	W-3-LP-OPS-REQ21	3

QUESTION

The plant is in Mode 5 preparing to perform refueling. Steam Generator manways are detensioned but not removed. RCS level is at midloop. The reactor was shutdown 96 hours ago. RCS temperature is 130°F. Determine the duration that SDC can be lost before entering Mode 4 conditions.

A. 7 minutes

B. 12 minutes

C. 15 minutes

D. 18 minutes

ANSWER

B

COMMENTS

Supply examinee with Attachments 2 and 3 of OP-901-131

Waterford 3 Examination Question
Examination Bank

Examination Question Number 26

Description:	Estimation of S/G tube leakage using charging and letdown mismatch			**Question ID:**	5673
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-202	2	1	6/23/99	CVC, RCS, PPO

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A37-AA2.04	3.4	3.7	W-3-LP-OPS-PPO20	3

QUESTION

A tube leak is in progress in S/G 1. The following conditions are noted:

- RCS Pressure is 2225 psia and rising
- PZR level is 55% and steady
- VCT level is 49% and lowering
- Letdown Flow = 30 GPM
- Charging Pumps A and AB are running
- RCP1A CBO = 1.5 GPM
- RCP1B CBO = 1.7 GPM
- RCP2A CBO = 1.5 GPM
- RCP2B CBO = 1.3 GPM

Assuming no additional leakage from the RCS to other sources, determine the leak rate into S/G 1.

A. 8 GPM

B. 14 GPM

C. 52 GPM

D. 58 GPM

ANSWER

C

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 27

Description:	Diagnose SGTR w/stuck open S/G safety			**Question ID:**	5674
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-009	0	0	9/9/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E38-EA2.02	4.5	4.8	W-3-LP-OPS-PPE01	12

QUESTION

After a reactor trip the following parameter values and trends are noted:

- PZR Level = 0%
- RCS Press = 1500 psia and lowering
- Loop 1 TC = 500°F and rapidly trending down
- Loop 2 TC = 530°F and slowly trending down
- Containment pressure = 14.9 psia and steady
- Containment Avg. Temp. = 105°F and lowering
- Containment area rad monitors = no alarms with readings trending down
- CVC-101, 103, and 109 are open
- RAB rad monitors = no alarms with steady readings
- MSL 1 rad monitor = alert alarm and trending up
- MSL 2 rad monitor = no alarm and steady
- S/G 1 WR level = 46% and dropping slowly
- S/G 2 WR level = 75% and steady

Assuming that all other equipment responded as designed, determine the events in progress.

A. MSLB and LOCA inside Containment

B. MSLB outside Containment and Letdown ISLOCA

C. Letdown ISLOCA and SGTR

D. MSLB outside containment and SGTR

ANSWER

D

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 28

Description:	Resetting Overspeed Trip and Manual Restart of EFW PUMP AB			**Question ID:**	5677
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-009-003	11	2	4/21/99	EFW, PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A54-AA1.02	4.4	4.4	W-3-LP-OPS-EFW00	13

QUESTION

A Loss of Main Feedwater is in progress with Emergency Feedwater restoring level to both Steam Generators when EFW Pump AB trips on electrical overspeed. The cause is determined to be improper draining of the steam line. Summarize the proper method for resetting and manually starting EFW Pump AB.

A. Reset the mechanical overspeed latch mechanism, depressurize the steam supply line, open the stop valve and verify EFW pump speed stabilizes between 4375 and 4450 RPM.

B. Close MS-401A and B, reset the mechanical overspeed latch mechanism, open the stop valve, open MS-401A and B, verify EFW pump speed stabilizes between 4375 and 4450 RPM.

C. Close MS-401A and B, depressurize the steam supply line, open the stop valve, open MS-401A and B, verify EFW pump speed stabilizes between 4375 and 4450 RPM.

D. Close MS-401A and B, depressurize the steam supply line, open MS-401A and B, open the stop valve, verify EFW pump speed stabilizes between 4375 and 4450 RPM.

ANSWER

C

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 29

Description:	Instrumentation/equipment affected by a loss of AB-DC			**Question ID:**	5676
Tier:	1	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-313	1	0	4/19/94	DC

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A58-AK1.01	2.8	3.1*	W-3-LP-OPS-PPO30	3

QUESTION

The AB DC bus has developed a short resulting in the loss of both AB battery chargers and the AB Battery. Which of the following is directly affected by the loss?

A. FWPT A and B Governor Valves

B. Main Turbine Trip Circuitry

C. FWPT A and B Trip Circuitry

D. CVC-209, Charging Header Isolation Valve

ANSWER

D

COMMENTS

Deleted

Examination Question Number 30

Description:	Interrelation between liquid release and DCT Rad Monitor			**Question ID:**	5681
Tier:	1	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-004-001	7	0	4/16/99	PPO, SP, PRM

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A59-AK2.01	2.7	2.8	W-3-LP-OPS-RMS00	2

QUESTION

The CCW system is operating with high activity due to leakage from a reactor coolant pump. DCT A develops a leak. How is an inadvertent release to offsite areas prevented?

A. DCT Sump 1 Radiation Monitor detects the high activity in the sump and closes the common discharge valve of the DCT Sump 1 pumps.

B. DCT Sump 1 Radiation Monitor detects the high activity in the sump and stops the DCT Sump 1 pumps.

C. DCT Sump 1 is normally aligned to the Liquid Waste System and only aligned to the storm sewers after a satisfactory sample is obtained.

D. DCT Sump 1 pump breakers are normally left open and are only closed after a satisfactory sample is obtained.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 31

Description:	Basis for selecting North or South CR Outside Air Intakes to provide makup air flow			**Question ID:**	5683
Tier:	1	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
OP-901-401	0	0	4/2/93

Plant Systems:
HVC, PPO

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A61-AK3.02	3.4	3.6	W-3-LP-OPS-PPO40	3

QUESTION

OP-901-401, High Airborne Activity in the Control Room has been entered due to an inadvertent release of a Waste Gas Decay Tank with a temperature inversion condition. The Control Room Supervisor orders the SNPO to determine which set of Control Room Outside Air Intakes (CROAI) should be opened for makeup to the Control Room envelope. What should the SNPO base this decision on?

A. The running Control Room Emergency Filtration Unit

B. The running Control Room Normal Air Handling Unit

C. The CROAI farthest from the Plant Stack

D. The lowest pair of CROAI radiation monitor readings

ANSWER

D

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 32

Description:	Prioritization of Safety Functions in OP-902-008			**Question ID:**	5684
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-008	11	0	9/9/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
2-4-6	3.1	4.0	W-3-LP-OPS-PPE08	6

QUESTION

OP-902-008, Safety Function Recovery Procedure has been implemented. Refer to the attached Safety Function Tracking Sheet and determine the priority for addressing the safety functions.

A. 1, 2, 3, 4, 5, 6, 7, 8, 9

B. 1, 7, 5, 2, 8, 3, 4, 6, 9

C. 1, 6, 2, 3, 7, 4, 5, 8, 9

D. 1, 7, 2, 3, 8, 4, 5, 6, 9

ANSWER

B

COMMENTS

Check (✔) the following on the Safety Function Tracking Sheet:
Success Path in Use – RC-3, MVA-DC-1, MVA-AC-2, IC-2, PC-1, HR-2, CI-1, CPTC-2, CCGC-1
SFSC Met – MVA-DC-1, MVA-AC-2, PC-1, CPTC-2, CCGC-1

Waterford 3 Examination Question
Examination Bank

Examination Question Number 33

Description:	Determination of Inadequate S/G Level to Throttle HPSI			**Question ID:**	5685
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-002	8	0	9/9/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E9-EK2.03	3.0	3.3	W-3-LP-OPS-PPE02	11

QUESTION

A small break LOCA is in progress and the following conditions are noted:

- PZR level is 10% and slowly rising
- PZR pressure is 570 psia and slowly rising
- RVLMS indicates 100% plenum level and 0% head lead level
- RCS Representative CET temperature is 450°F and slowly lowering
- S/G 1 level is 45% NR and dropping slowly
- S/G 2 level is 48% NR and steady
- EFW pumps are running, EFW Flow Controllers are in manual

To throttle HPSI flow which of the following is true?

A. Level in at least one S/G must be raised.

B. Reactor Vessel level must be raised.

C. PZR level must be raised.

D. RCS Subcool Margin must be raised.

ANSWER

A

COMMENTS

Supply examinees with copy of steam tables

Examination Question Number 34

Description:	Manual Restoration of Power to Startup Channel			**Question ID:**	5686
Tier:	1	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-500-008	11	0	12/10/98	ENI, PPR

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A32-AA1.01	3.1*	3.4*	W-3-LP-OPS-ENI00	4

QUESTION

The plant is at 2 E-07% reactor power after a reactor trip when ENI Safety Channel B middle detector fails high. The PNPO notes that Startup Channel 2 has de-energized and the Startup Channel 2 Offnormal annunciator is flashing. What is the recommended action to restore power to the startup channel?

A. De-energize Safety Channel B ENI drawer at CP-10.

B. De-energize Safety Channel B ENI drawer at Sups MB.

C. Place the HV Control Switch to Alternate in Startup Channel 2 drawer.

D. Place the HV Control Switch to Primary in Startup Channel 2 drawer.

ANSWER

D

COMMENTS

Examination Question Number 35

Description:	PZR tailpipe temperature of open PZR Safety Valve			**Question ID:**	5687
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	Modified Bank (used)	**History:**	1997 SRO NRC exam (previous version)		

Reference	Revision	Change	Date	Plant Systems:
CE Steam Table Mollier Diagram				PPE, RCS

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A8-AK1.01	3.2	3.7	W-3-LP-OPS-TYH04	21

QUESTION

Given the following plant conditions and assumptions:

- A PZR Safety Valve is stuck open
- PZR pressure is 500 psia and steady
- Quench Tank pressure is 15 psig and steady
- Quench Tank Temperature is 200°F
- The reactor has tripped
- Assume ambient heat losses are negligible and the steam quality in the pressurizer is 100%

Determine the expected temperature indicated downstream of the open PZR Safety Valve.

A. 200°F

B. 250°F

C. 330°F

D. 465°F

ANSWER

C

COMMENTS

Modified from 1997 Waterford III SRO examination
Supply examinees with copy of CE Steam tables with Mollier Diagram

Waterford 3 Examination Question
Examination Bank

Examination Question Number 36

Description:	Actions in EOPs associated with an ATWS			**Question ID:**	5688
Tier:	1	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-000	8	0	9/8/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E29-EK3.12	4.4	4.7	W-3-LP-OPS-CVC00	3
			W-3-LP-OPS-PPO10	4
			W-3-LP-OPS-PPE01	8

QUESTION

OP-902-000 requires initiation of emergency boration if more than one CEA is not fully inserted. What is the basis for closing the VCT Outlet Valve, CVC-183, during this operation?

A. To ensure only the Boric Acid Makeup Tanks provide flow to the suction of the Charging pumps.

B. To prevent filling the VCT with water from the Boric Acid Makeup Tanks.

C. To prevent losing adequate NPSH to the Charging pumps as VCT level lowers.

D. To prevent filling the Boric Acid Makeup Tanks with water from the VCT.

ANSWER

A

COMMENTS

Examination Question Number 37

Description:	Actions required for failed open Steam Bypass Valve and SGTR during OP-902-000			**Question ID:**	5689
Tier:	1	**Group:**	2	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-000	8	0	9/8/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E7-EA1.03	4.2	4.1	W-3-LP-OPS-PPE01	10

QUESTION

A manual reactor trip is initiated in accordance with OP-901-202, Steam Generator Tube Leakage or High Activity. During the performance of Standard Post Trip Actions the PNPO notes that RCS T_C is 532°F and lowering. The SNPO notes that one steam bypass valve controller is failed at 100% output. What is the preferred action to be taken in this situation?

A. Allow RCS temperature to continue to lower until T_H reaches 520°F, then take manual control of steam bypass control valves to maintain T_H less than 520°F.

B. Take manual control of steam bypass control valves and restore T_C to between 535°F and 555°F.

C. Manually initiate a Main Steam Isolation Signal and verify the Atmospheric Dump Valves control T_C between 535°F and 555°F.

D. Manually close the Main Steam Isolation Valves and verify the Atmospheric Dump Valves control T_C between 535°F and 555°F.

ANSWER

B

COMMENTS

Examination Question Number 38

Description:	Requirement to trip reactor for excessive RCS leakage			**Question ID:**	5690
Tier:	1	**Group:**	3	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-111	1	0	10/15/97	PPO

NRC KA Number	RO	SRO	Training Material	Objective
2-4-4	4	4.3	W-3-LP-OPS-PPO10	3

QUESTION

Given the following conditions:

- OP-901-111, RCS Leak has been entered due to excessive RCS leakage
- RCS Tavg is 573°F and steady
- AB charging pump is OOS
- Charging pumps A and B are running
- Letdown secured to determine location of RCS leak
- RCS Leakage = 90 gpm

Which of the following is the appropriate action to be performed in accordance with OP-901-111?

A. Maintain RCS conditions stable to prevent loss of PZR level, while attempting to locate the leak.

B. Commence a normal shutdown in accordance with OP-010-005, Plant Shutdown, and be in Mode 3 within 6 hours.

C. Commence a rapid plant shutdown in accordance with OP-901-212, Rapid Plant Power Reduction and be in Mode 3 within 1 hour.

D. Perform a manual reactor trip, manually initiate SIAS and CIAS, and go to OP-902-000, Standard Post Trip Actions.

ANSWER

D

COMMENTS

Examination Question Number 39

Description:	Behavior of IA pressure following a LOOP			**Question ID:**	5691
Tier:	1	**Group:**	3	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
W-3-LP-OPS-AIR00				IA

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A56-AA1.37	3.4	3.5	W-3-LP-OPS-AIR00	1

QUESTION

Given the following initial conditions:

- IA compressor A is running; IA compressor B is in standby
- Four CW pumps are running
- TC Pump A is running
- SA compressors B and C are running
- SA to IA Cross-connect Valve setpoint is 100 psig
- Instrument Air and Station Air usage are within the capacity of one air compressor
- SUT A and B are supplying in-house loads

Determine the response of the Instrument Air system after the actuation of SUT A Lockout Relay 87STA.

A. IA pressure drops until action is taken to manually cross-connect Instrument Air and Station Air, then IA pressure recovers.

B. IA pressure drops until IA compressor B starts automatically, then IA pressure recovers.

C. IA pressure continues to drop until alternate cooling is aligned to IA compressor B and the compressor is started manually, then IA pressure recovers.

D. IA pressure drops until Station Air and Instrument air are automatically cross-connected, then IA pressure varies around the setpoint of the automatic valve.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 40

Description: Affect on CVCS when PZR level controller fails low

Question ID: 5692

Tier: 1 **Group:** 3

Cognitive Level: Memory/Fund

Source: New **History:** N/A

Reference	Revision	Change	Date
OP-901-110	03	01	9/2/98

Plant Systems:
PLC, PPO

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A28-AK2.03	2.6	2.9	W-3-LP-OPS-CVC00	3

QUESTION

The PZR level controller on CP-2 failed low with the letdown flow controller in Auto. The letdown flow controller is affected in the following way:

A. The letdown flow controller fails to 0% output.

B. The letdown flow controller fails to 100% output.

C. The letdown flow controller fails to a preset minimum output.

D. The letdown flow controller fails to a preset maximum output.

ANSWER

C

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 41

Description:	CEA position information Reed Switch Vs. Pulse Counter			**Question ID:**	5693
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
Sys Desc for CED			

Plant Systems:
CED

NRC KA Number	RO	SRO	Training Material	Objective
3.1-001-A3.02	3.7	3.6	W-3-LP-OPS-CED00	4
			W-3-LP-OPS-CPC00	4

QUESTION

All of the following provide the capability to monitor actual position for an individual CEA using magnetic reed switches **EXCEPT** the:

A. CEA Position Display CRT on CP-2

B. Digital CEA INDV indicator on CP-2

C. CEAC Channels 1 and 2 on CP-7

D. Targeted CPC Channel on CP-7

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 42

Description:	Determine the amount of Boron required to withdraw Group 6 CEAs out			**Question ID:**	5694
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-002-005	14	8	2/21/99	CED

NRC KA Number	RO	SRO	Training Material	Objective
3.1-001-A4.05	3.7	3.7	W-3-LP-OPS-CVC00	11

QUESTION

Given the following conditions:

- The plant has been operating at 100% power for 275 days
- Reg Group 6 CEAs are inserted to 120" for ASI control and ASI is moving to the bottom of the core
- Tavg = 573°
- RCS Boron Concentration = 400 ppm
- Core Burnup = 450 EFPD
- Boric Acid Makeup Tank (BAMT) A and B Boron Concentration = 5500 ppm

Determine the number of gallons of Boric Acid that must be added to the RCS from the BAMTs to withdraw Reg Group 6 CEAs to 150".

A. 122 gallons

B. 138 gallons

C. 144 gallons

D. 244 gallons

ANSWER

A ~~B~~ B

COMMENTS

Supply examinees with Plant Data Book curves 1.4.3.1-1.4.5.2, 1.5.2.1, 1.5.2.3, 1.5.3.1, 1.5.3.3, 1.5.4.1, and 1.5.4.3
Supply examinees with copy of OP-002-005, Attachment 11.6

Waterford 3 Examination Question
Examination Bank

Examination Question Number 43

Description:	Determine the effect of starting a RCP with a large primary to secondary delta-T			**Question ID:**	5695
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
TS 3.4.8				RCP, TS, RCS

NRC KA Number	RO	SRO	Training Material	Objective
3.4-003-A1.07	3.4*	3.4	W-3-LP-OPS-RCP00	9, 13
			W-3-LP-OPS-RCS00	9, 11

QUESTION

The following conditions exist:

- RCS is filled and vented and the PZR is solid
- RCS pressure is being controlled by both letdown backpressure control valves
- RCS pressure = 370 psia
- CET temperature = 130°F
- Steam Generator 1 water temperature = 220°F
- All RCPs are secured

Which of the following is true when starting RCP 1A under these conditions.

A. RCS pressure would rise to the setpoint of the LTOP reliefs; the combined capacity of both LTOP reliefs is required to protect the RCS from overpressurization.

B. RCS pressure would lower but be controlled by the letdown backpressure control valves above the minimum pressure for running the RCP.

C. RCS pressure would rise above the setpoint of the LTOP reliefs; the RCS is protected from overpressurization by the capacity of one LTOP relief.

D. RCS pressure would drop below the pressure for operating a RCP causing cavitation of the RCP and possible damage to the impeller and seals.

ANSWER

C

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 44

Description:	Effect of VCT level transmitter failing low			**Question ID:**	5696
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-002-005	1	0	2/21/9	CVC
OP-901-113	01	00	2/2/98	

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.2-004-K4.12	3.1	3.4	W-3-LP-OPS-CVC00	3

QUESTION

VCT level transmitter CVC-ILT-0227 fails low. Select the effect of this failure on the CVC system.

A. A continuous automatic makeup demand signal would be present; high and low level alarms are disabled.

B. Automatic diversion of the VCT inlet valve on high level is disabled; Charging pump suction automatically aligns to the RWSP and the VCT suction path isolates.

C. A continuous automatic makeup demand signal would be present; Charging pump suction automatically aligns to the RWSP and the VCT suction path isolates.

D. Automatic diversion of the VCT inlet valve on high level is disabled; high and low level alarms are disabled.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 45

Description:	Limiting conditions for operations and safety limits for RCPs			**Question ID:**	5697
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
TS 3.4.1			

Plant Systems:
RCP, RCS

NRC KA Number	RO	SRO	Training Material	Objective
2-2-22	3.4	4.1	W-3-LP-OPS-RCP00	8

QUESTION

All of the following are true concerning the OPERABILITY of reactor coolant loops in MODE 3 **EXCEPT**:

A. Only one RCP must be OPERABLE in each loop.

B. OPERABILITY of the required RCPs is verified by correct breaker alignment and power availability.

C. Only one RCP is required to be in operation to satisfy the requirements of the LCO.

D. Securing all RCPs is **NOT** allowed by the LCO.

ANSWER

D

COMMENTS

Examination Question Number 46

Description:	Interrelations between CVCS and PLCS			**Question ID:**	5698
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
W-3-LP-OPS-PLC00				CVC

NRC KA Number	RO	SRO	Training Material	Objective
3.1-004-K1.01	3.6	4	W-3-LP-OPS-PLC00	1

QUESTION

Given the following initial conditions:

- The plant is at 100% power
- PZR level and Tavg are at programmed setpoint
- All systems are operating in automatic

Governor Valve #1 fails closed. Describe the response of the PLC and CVC systems.

A. The PZR level controller output rises, the letdown flow control valves throttle open to raise letdown flow, and letdown backpressure control valves throttle open to return letdown pressure to setpoint.

B. The PZR level controller output rises, the letdown flow control valves throttle open to raise letdown flow, and letdown backpressure control valves throttle closed to return letdown pressure to setpoint.

C. The PZR level controller output lowers, the letdown flow control valves throttle closed to lower letdown flow, and the letdown backpressure control valves throttle open to return letdown pressure to setpoint.

D. The PZR level controller output lowers, the letdown flow control valves throttle closed to lower letdown flow, and the letdown backpressure control valves throttle closed to return letdown pressure to setpoint.

ANSWER

A

COMMENTS

Examination Question Number 47

Description:	Initiation signals for ESF circuit logic			**Question ID:**	5699
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
TS 3.3.2				PPS

NRC KA Number	RO	SRO	Training Material	Objective
3.2-013-K1.01	4.2	4.4	W-3-LP-OPS-CC00	3

QUESTION

Given the following initial conditions:

- Containment Pressure = 17.6 psia
- PZR Pressure = 1700 psia
- SG 1 Pressure = 700 psia
- SG 2 Pressure = 800 psia
- SG 1 NR Level = 0%
- SG 2 NR Level = 17%

Which of the following components would not receive an automatic initiation signal as a result of the above parameters?

A. SI-139B, Cold Leg 1B LPSI Flow Control Valve

B. EFW-229B, SG 2 Backup EFW Isolation Valve

C. CC-963A, SDC HX A CCW Outlet Flow Control Valve

D. FW-166A, SG 1 Startup Feedwater Regulating Valve

ANSWER

C

COMMENTS

Examination Question Number 48

Description:	RCP seal failure and subsequent shutdown			**Question ID:**	5712
Tier:	1	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-130	02	00	4/6/98	RCP

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A15/17-AK1.02	3.7	4.1	W-3-LP-OPS-PPO10	3,4
			W-3-LP-OPS-RCP00	13

QUESTION

You are the dayshift PNPO. Shortly after relieving the shift, the lower seal fails on Reactor Coolant Pump 1A. The decision is made to continue operation with increased monitoring for further degradation. At 10:30 RCP 1A seal pressures are as follows:

Vapor	0 psia
Upper	1000 psia
Middle	2250 psia

Which of the following are appropriate?

A. The upper seal has failed, trip the reactor, trip RCP 1A, and perform Standard Post Trip actions.

B. The upper seal has failed, commence a rapid plant shutdown IAW OP-902-212.

C. The vapor seal has failed, commence a rapid plant shutdown IAW OP-902-212.

D. The vapor seal has failed, trip the reactor, trip RCP 1A, and perform Standard Post Trip actions.

ANSWER

D. The vapor seal has failed, trip the reactor, trip RCP 1A, and perform Standard Post Trip actions.

COMMENTS

Examination Question Number 49

Description:	Operational Characteristics of natural circ			**Question ID:**	5713
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-002	8	0	9/9/98	PPE, SI, RCS

NRC KA Number	RO	SRO	Training Material	Objective
4.4-A13-AK2.2	3.4	3.6	W-3-LP-OPS-PPE05	1

QUESTION

All of the following represent operational characteristics important for natural circulation **EXCEPT**:

A. controlling Atmospheric Dump Valves at or below saturation pressure for T_{COLD}
B. maintaining RCS pressure IAW 28 deg subcooled curve
C. maintaining Safety Injection flow IAW flow curves
D. initiating hot and cold leg injection 2-4 hours post LOCA

ANSWER

D. initiating hot and cold leg injection 2-4 hours post LOCA

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 50

Description:	Emergency boration on loss of 3A bus			**Question ID:**	5708
Tier:	1	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**
OP-901-103	00	01	9/22/95
OP-902-000	08	00	9/8/98

Plant Systems:
CVC, PPE

NRC KA Number	**RO**	**SRO**
4.2-A24-AA2.02	3.9	4.4

Training Material	**Objective**
W-3-LP-OPS-PPE01	9
W-3-LP-OPS-PPO10	2

QUESTION

The reactor has just tripped due to a loss of off site power. Both Emergency Diesel Generators started as required, but EDG A's output breaker failed to close. As the PNPO, what are your required actions when you notice 2 CEAs stuck out?

A. Commence Emergency Boration using both Gravity Feed valve A & B.
B. Commence Emergency Boration using BAM Pump B and Gravity feed valve B.
C. Commence Emergency Boration using both BAM Pump A & B
D. Commence Emergency Boration using BAM Pump A and Gravity Feed valve A.

ANSWER

A. Commence Emergency Boration using both Gravity Feed valve A & B.

COMMENTS

Examination Question Number 51

Description:	CCW leak and Essential Chillers			**Question ID:**	5710
Tier:	1	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-510	03	00	6/2/98	CC, PPO, CHW

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A26-AA1.02	3.2	3.3	W-3-LP-OPS-CC00	3,7
			W-3-LP-OPS-PPO10	4

QUESTION

The plant is operating at 100% when a 1300 gpm leak develops on the supply piping to Shutdown Cooling Heat Exchanger A. The CCW System responds properly to the failure. CCW Surge Tank level for train A is 0%, train B is 55%. Which of the following components still has the means available to perform its design function?

A. Containment Fan Cooler C
B. Essential Chiller A
C. Emergency Diesel Generator A
D. High Pressure Safety Injection Pump A

ANSWER

B. Essential Chiller A

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 52

Description:	Pressurizer Pressure Controller failure without action			**Question ID:**	5714
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-901-120	02	00	12/8/97	PPC, PPS
OP-004-006	13	00	10/19/98	

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
4.2-A27-AK2.03	2.6	2.8	W-3-LP-OPS-PLC00	5,7

QUESTION

The plant is at 100% power when the output on Pressurizer Pressure controller, RC-IPIC-0100 fails high. Which response describes why the reactor trips if no operator action is taken?

A. Low pressure trip from the Core Protection Calculators.
B. Low pressure trip from the Plant Protection System.
C. High pressure trip from the Core Protection Calculators.
D. High pressure trip from the Plant Protection System.

ANSWER

A. Low pressure trip from the Core Protection Calculators.

COMMENTS

Examination Question Number 53

Description:	What remains energized on loss of 2A bus			**Question ID:**	5743
Tier:	2	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-003-016	8	4	7/13/99	ED

NRC KA Number	RO	SRO	Training Material	Objective
3.6-062-K2.01	3.3	3.4	W-3-LP-OPS-AIR00	3

QUESTION

The plant is in mode 3 with preparation being made for a reactor startup when the Startup Transformer feeder to the 2A bus trips open. All automatic actions occurred as designed. Which of the following components will still have power available?

A. Reactor Drain Tank Pump

B. Instrument Air Compressor A

C. Blowdown Pump A

D. Containment Spray Riser Level Pump A

ANSWER

B. Instrument Air Compressor A

COMMENTS

Examination Question Number 54

Description:	Excess steam demand and PTS			**Question ID:**	5735
Tier:	1	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-009	0	0	9/9/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.4-A11-AK1.1	3.1	3.3	W-3-LP-OPS-PPE04	4

QUESTION

Plant conditions are as follows:

- RCS pressure 1100 psia, slowly lowering
- T_{CET} 465 ° F, slowly lowering
- Containment pressure 27.45 psia
- Pressurizer level 0 %
- SG #1 pressure 375 psia, 35% WR
- SG #2 pressure 725 psia, 60% WR
- Containment temp 200 ° F

Which of the following describes the correct actions for this event?

A. When RCS pressure turns, stabilize pressure greater than HPSI shutoff head using Pressurizer spray.

B. When T_{CET} starts to rise, fully open 1 Steam Bypass valve to stabilize RCS temperature.

C. Manually initiate EFAS for SG #1 and commence feeding in manual to restore level in SG #1.

D. Immediately open Aux. Spray valves to lower RCS pressure and maximize HPSI flow to restore Pressurizer level.

ANSWER

A. When RCS pressure turns, stabilize pressure greater than HPSI shutoff head using Pressurizer spray.

COMMENTS

Examination Question Number 55

Description: Loss of Condenser vacuum and Steam Bypass

Question ID: 5718

Tier: 1 **Group:** 1

Cognitive Level: Memory/Fund

Source: New **History:** N/A

Reference	Revision	Change	Date
OP-901-220	2	0	7/31/97

Plant Systems:
PPO, SBC

NRC KA Number	RO	SRO
4.2-A51-AK3.01	2.8*	3.1*

Training Material	Objective
W-3-LP-OPS-PPO20	3
W-3-LP-OPS-SBC00	2

QUESTION

A plant downpower is in progress due to lowering Condenser vacuum. The shift is implementing OP-901-220, Loss of Condenser Vacuum along with OP-010-005, Plant Shutdown. Which of the following Main Condenser vacuum parameters correctly describes the associated setpoints and actuations?

A 23 INHG, Main Turbine trips.

B 20 INHG, Main Feed Pumps trip.

C 14 INHG, Condenser Vacuum Pumps shift to hogging.

D 3.4 INHG, Steam Bypass valves close.

ANSWER

D 3.4 INHG, Steam Bypass valves close.

COMMENTS

Examination Question Number 56

Description:	Isolating CCW to the RCP on Station Blackout			**Question ID:**	5734
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-005	10	0	9/9/98	CC, RCP, PPE

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E55-EK3.02	4.3	4.6	W-3-LP-OPS-PPE05	3,7

QUESTION

OP-902-005, Station Blackout Recovery, directs the operator to verify closed CC-641, CC-710, and CC-713 to isolate Component Cooling Water to Containment. What is the reason for this?

A. To prevent thermal shocking the CEDM Coolers upon restart of the Component Cooling Water Pumps.
B. To prevent water hammer in Containment Bldg. piping upon restart of the Component Cooling Water Pumps.
C. To prevent thermal shocking the RCPs upon restart of the Component Cooling Water Pumps.
D. To prevent runout of the Component Cooling Water Pumps upon restart.

ANSWER

C. To prevent thermal shocking the RCPs upon restart of the Component Cooling Water Pumps.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 57

Description:	Sequence of events following fault on 3BS load circuit (new SUPS)			**Question ID:**	5528
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	Bank	**History:**	Not Used		

Reference	Revision	Change	Date	Plant Systems:
OP-006-005	10	0	3/12/99	ID

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A57-AA2.17	3.1	3.4	W-3-LP-OPS-ID00	1
3.6-062-A2.02	2.2	2.6	W-3-LP-OPS-RF00	3

Deleted

QUESTION

A 100% short to ground on the circuit to CVC-103 has occurred. The power supply is from a circuit on PDP 91B, which is supplied from 3BS vital SUPS. Which of the following sequence of events will occur when the SUPS senses the fault?

A. The SUPS will automatically shut down.
B. The transfer switch will automatically transfer to the alternate supply, the circuit breaker for supply to CVC-103 will open on overload, and the transfer switch will return to the normal source.
C. The transfer switch will automatically transfer to the alternate supply, the circuit breaker for supply to CVC-103 will open on overload. The SUPS can then be returned to normal source by the operator using the "inverter to normal" pushbutton.
D. The circuit breaker supplying CVC-103 will open and the SUPS will remain in service.

ANSWER

B. The transfer switch will automatically transfer to the alternate supply, the circuit breaker for supply to CVC-103 will open on overload, and the transfer switch will return to the normal source

COMMENTS

DCP-3546 (RF09)

Examination Question Number 58

Description:	ACC-126 failure, effects on CCW Surge Tank			**Question ID:**	5736
Tier:	1	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
Sys Desc for CC			

Plant Systems:
ACC, CC

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A62-AA1.03	3.6*	3.6	W-3-LP-OPS-CC00	3

QUESTION

Plant conditions are as follows:

CCW Surge Tank level is 85 % and rising. CCW radiation monitors on train A and B are reading normal. Outside air temperature is 88 °F, and ACCW trains A and B are in operation. Which selection describes the event in progress?

A. A leak has developed in the Letdown Heat Exchanger.
B. A leak has developed in the EDG A Jacket Water cooler.
C. Train A Dry Cooling Tower Fan Sequencer has failed, driving all fans to fast.
D. ACC-126 A m/a station has failed to manual with 0 output.

ANSWER

D. ACC-126 A m/a station has failed to manual with 0 output.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 59

Description:	Emergency Boration during CR evacuation			**Question ID:**	4124 N
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	Used Bank	**History:**	1996 RO NRC exam		

Reference	Revision	Change	Date	Plant Systems:
OP-901-502	5	1	7/8/99	PPO

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A68-AK3.18	4.2	4.5	W-3-LP-OPS-PPO51	08
4.2-A67-AK3.04	3.3	4.1		

QUESTION

A Control Room evacuation has been ordered. The PNPO was not able to trip the reactor prior to evacuation. Why is emergency boration required?

A. The Main Turbine will not trip when the reactor is tripped outside the control room.

B. The MSR temperature control valve logic cannot be reset from outside the control room.

C. The Feedwater Control System will not shift to the RTO mode when the reactor is tripped from outside the control room.

D. The operator is unable to verify CEA position following a trip from outside the control room.

ANSWER

D. The operator is unable to verify CEA position following a trip from outside the control room.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 60

Description: Evacuation of Control Room, Control of heat sink

Question ID: 5711

Tier: 1 **Group:** 1

Cognitive Level: Comp/Analysis

Source: New **History:** N/A

Reference	Revision	Change	Date
OP-901-502	5	1	7/8/99

Plant Systems:
EFW, PPO, MS

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A68-AA2.05	4.2	4.3	W-3-LP-OPS-PPO51	20, 21

QUESTION

A Control Room evacuation has been ordered. Which of the following describe initial plant cooldown?

A. Controlling Steam Bypass valves, feeding with the Auxiliary Feed Pump

B. Controlling Steam Bypass valves, feeding with Emergency Feedwater

C. Controlling Atmospheric Dump valve, feeding with the Auxiliary Feed Pump

D. Controlling Atmospheric Dump valve, feeding with Emergency Feedwater.

ANSWER

D. Controlling Atmospheric Dump valve, feeding with Emergency Feedwater.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 61

Description:	Override required for Containment Fan Cooler CCW isolations			**Question ID:**	5709
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-902-009	0	0	9/9/98	CC, CCS

NRC KA Number	RO	SRO	Training Material	Objective
2-4-35	3.3	3.5	W-3-LP-OPS-CCS00	11
			W-3-LP-OPS-PPE01	6

QUESTION

A LOCA inside containment has occurred and OP-902-002 is being implemented. Containment Fan Cooler A trips on electrical overload. Which actions are required to satisfy Containment Isolation?

A. Place the keyswitch on Aux Panel 1 to OVERRIDE, take Containment Fan Cooler A control switch to STOP, and verify CC-808 A and CC-822 A close.
B. Manually gag CC-808 A and CC-822 A closed.
C. Place the keyswitch on Aux Panel 1 to OVERRIDE and verify CC-808 A and CC-822 A close.
D. Place the keyswitch in the –4 wing area to OVERRIDE and verify CC-808 A and CC-822 A close.

ANSWER

C. Place the keyswitch on Aux Panel 1 to OVERRIDE and verify CC-808 A and CC-822 A close.

COMMENTS

Examination Question Number 62

Description:	Use of Startup Channels for RV level determination			**Question ID:**	27
Tier:	1	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	Bank	**History:**	Not Used		

Reference	Revision	Change	Date
W-3-LP-OPS-MCD03			

Plant Systems:
ENI, MCD

NRC KA Number	RO	SRO	Training Material	Objective
4.1-E74-EK1.02	4.6	4.8	W-3-LP-OPS-MCD03	6

QUESTION

The plant has undergone a large Loss of Coolant Accident. The Reactor Coolant Pumps are secured. There is no safety injection flow. The reactor coolant in the core region is at saturation conditions.

SELECT the expected response of the Excore Nuclear Instrumentation System Startup Channels as the coolant in the core region is initially boiled off.

A. As level in the core drops the level in the down comer drops causing count rate to Rise.

B. As level in the core drops the level in the down comer rises causing count rate to Lower.

C. As level in the core drops there are fewer neutrons generated causing count rate to Lower.

D. As level in the core drops there are more neutrons generated causing count rate to Rise.

ANSWER

a

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 63

Description:	Chemistry dept. responsibilities for high RCS activity			**Question ID:**	4125 N
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	Used Bank	**History:**	1996 RO NRC exam		

Reference	Revision	Change	Date	Plant Systems:
OP-901-410	0	0	4/16/93	PPO, RCS

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A76-AK3.06	3.2	3.8	W-3-LP-OPS-PPO40	03

QUESTION

All of the following items are sampled and analyzed by the W-3 Chemistry Department when OP-901-410, High Activity in Reactor Coolant System is **EXCEPT**:

A. Cesium Concentration

B. Dose Equivalent Iodine Concentration

C. Dose Equivalent Iodine isotopic analysis

D. Gross RCS activity

ANSWER

A. Cesium Concentration

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 64

Description:	Shift turnover and board walkdowns			**Question ID:**	5721
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-100-007	13	1	5/25/99	ADM

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
2-1-3	3.0	3.4	w-3-lp-ops-ppa00	2

QUESTION

All of the following are true regarding board walkdowns **EXCEPT**:

A. Each setpoint indicator, switch, and controller should be checked.
B. The PNPO and SNPO should walk down their boards twice each shift.
C. Prior to removing a safety related component from service, each bypass status indicating lamp, controller, and switch should be checked on the opposite train component.
D. A walkdown should be performed in the last 2 hours of the shift on any safety-related components that were manipulated during the shift.

ANSWER

B. The PNPO and SNPO should walk down their boards twice each shift.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 65

Description:	Safety limits and tech spec 6.7.1			**Question ID:**	5722
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**
TS 2.1.2			
TS 2.1.1			

Plant Systems:
ADM

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
2-1-11	3	3.8	W-3-LP-OPS-TS00	3
			W-3-LP-OPS-TS03	10

QUESTION

All of the following prohibit critical operation of the plant until authorized by the NRC **EXCEPT**:

A. DNBR goes below 1.26 while in mode 1
B. Reactor Coolant pressure exceeds 2750 psia while in mode 3
C. Post trip analysis shows linear power peaked at 120% rated thermal power
D. Peak linear heat rate exceeds 21.0 kW/ft while in mode1.

ANSWER

C. Post trip analysis shows linear power peaked at 120% rated thermal power

COMMENTS

Examination Question Number 66

Description:	Plant page requirements			**Question ID:**	5724
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-100-001	16	0	2/19/99	ADM, COM

NRC KA Number	RO	SRO	Training Material	Objective
2-1-16	2.9	2.8	W-3-LP-OPS-PPA00	2

QUESTION

Plant Page announcements concerning operation of major plant equipment

A. Should be made twice without an operator in attendance at the piece of equipment, once with an operator in attendance.
B. Should not be made if an operator is in attendance at the piece of equipment.
C. Should be made once without an operator in attendance at the piece of equipment, twice with an operator in attendance.
D. Should always be made only once.

ANSWER

A. Should be made twice without an operator in attendance at the piece of equipment, once with an operator in attendance.

COMMENTS

Examination Question Number 67

Description:	Frequency for updating field control verifications			**Question ID:**	1630
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	Modified Bank	**History:**	Not Used		

Reference	Revision	Change	Date	Plant Systems:
OP-100-001	16	00	2/19/99	PPA
W5.201	3	1	9/15/97	

NRC KA Number	RO	SRO	Training Material	Objective
2-1-21	3.1	3.2	W-3-LP-OPS-PPA01	1

QUESTION

How long is a Field-Controlled copy of an Operation's procedure allowed to be used before reverification is required?

A. 7 days
B. 14 days
C. 30 days
D. 60 days

ANSWER

B. 14 days

COMMENTS

Examination Question Number 68

Description:	Criticallity below PDIL			**Question ID:**	4548
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	Bank	**History:**	Not Used		

Reference	Revision	Change	Date
OP-010-003	0	1	3/22/99

Plant Systems:
CED, PPO, PPN

NRC KA Number	RO	SRO
2-1-20	4.3	4.2
2-2-1	3.7	3.6

Training Material	Objective
W-3-LP-OPS-PPN01	3

QUESTION

While performing a reactor startup, criticality is expected with Group 4 Control Rods below 90 inches. Which one of the following actions must be taken as a result of this?

A. Fully insert all CEAs, verify boron concentration, and recalculate ECC.
B. Commence Emergency Boration, fully insert all CEAs, and verify boron concentration.
C. Fully insert all Reg Group CEAs, verify boron concentration, and recalculate ECC.
D. Commence Emergency Boration, fully insert all Reg Group CEAs, and verify boron concentration.

ANSWER

B

COMMENTS

Examination Question Number 69

Description:	UNT-005-003, Clearance Procedure			**Question ID:**	5385
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	Bank	**History:**	Not Used		

Reference	Revision	Change	Date	Plant Systems:
UNT-005-003	16	0	12/1/98	ADM

NRC KA Number	RO	SRO	Training Material	Objective
2-2-13	3.6	3.8	W-3-LP-OPS-CLR00	04

QUESTION

A Craft Maintenance Tagout

A. Allows several maintenance personnel to operate the tagged component.
B. On Safety Equipment requires an operator to verify tags are hanging correctly.
C. Will use normal red danger tags with CM in the identifier.
D. Requires the holder to attach a label identifying the name of the holder to each craft maintenance tag listed on the craft maintenance tagout.

ANSWER

D.

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 70

Description:	Tech Spec 4.0.3			**Question ID:**	5729
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
TS 3.5.1				ADM
TS 4.0.3				

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
2-2-22	3.4	4.1	W-3-LP-OPS-TS02	8, 9

QUESTION

Chemistry Department reported to the Control Room at 1000 on 7/20/99 that the monthly sample for SIT 2A was performed incorrectly and the results are invalid. The late date for the sample was 1430 on 7/11/99. What are the required actions?

A. Declare SIT 2A inoperable and enter tech spec 3.5.1 at 1000 on 7/20/99. Invoke tech spec 4.0.3, and complete the required sample within 24 hours.

B. Declare SIT 2A inoperable and enter tech spec 3.5.1 as of 1430 on 7/11/99. Invoke tech spec 4.0.3 and complete the required sample within 24 hours.

C. Declare SIT 2A inoperable and enter tech spec 3.5.1 as of 1430 on 7/11/99. In the next 6 hours either be in Hot Shutdown or complete the required surveillance.

D. Declare SIT 2A inoperable and enter tech spec 3.5.1 at 1000 on 7/20/99. Declare SIT 2A operable by completing the required surveillance within 1 hour or be in Hot Shutdown in the next 6 hours.

ANSWER

A. Declare SIT 2A inoperable and enter tech spec 3.5.1 at 1000 on 7/20/99. Invoke tech spec 4.0.3, and complete the required sample within 24 hours.

COMMENTS

Provide tech spec 3.5.1

Waterford 3 Examination Question
Examination Bank

Examination Question Number 71

Description:	Very high and locked high radiation areas			**Question ID:**	5730
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
HP-001-107	12	2	6/11/98	RAD

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
2-3-1	2.6	3	W-3-LP-OPS-PPA00	2
			W-3-LP-OPS-RAD02	6

QUESTION

A Locked High Radiation Area is an area accessible to individuals in which radiation levels could result in an individual receiving a DDE ≥ _____ in 1 hour at 30 cm from the radiation source or any surface that radiation penetrates, and a Very High Radiation Area is an area where an individual would receive a DDE ≥ _____ in 1 hour at ~~30 cm.~~ 1 meter 10/1/99

A. 500 rads, 1 rem
B. 1 rem, 500 rads
C. 100 mrem, 1 rem
D. 5 rem, 500 rads

ANSWER

B. 1 rem, 500 rads

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 72

Description:	Power entries with HP-001-213			**Question ID:**	5731
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
HP-001-213	9	2	6/11/98	ADM, RAD

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
2-3-10	2.9	3.3	W-3-LP-OPS-PPA00	2

QUESTION

Which of the following is true regarding Containment building entries while at power?

A. Entry into the Safety Injection Sump and Containment Sump is allowed.
B. A health physics technician escort is not required if a pre-job brief was held.
C. The GMPO permission is required for non-emergency power entries
D. Reactor power shall be held stable, with < 3% power rise per hour.

ANSWER

D. Reactor power shall be held stable, with < 3% power rise per hour.

COMMENTS

Examination Question Number 73

Description:	Immediate operator action for SDC Malfunction.			**Question ID:**	1195
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Memory/Fund
Source:	Used Bank	**History:**	RO class 2-3-99		

Reference	Revision	Change	Date	Plant Systems:
OP-901-131	1	1	2/9/99	PPO, SDC

NRC KA Number	RO	SRO	Training Material	Objective
4.2-A25-AA1.03	3.4	3.3	W-3-LP-OPS-REQ21	2
2-4-1	4.3	4.6		

QUESTION

The following plant conditions exist:

- Refueling Outage
- Mode 6
- Shutdown Cooling Trains A&B in service

WHICH of the following conditions requires the operator to secure LPSI Pump B as an Immediate Operator Action?

A. SI-407 B, Loop 1 Suction Isol Downstream Outside Valve, closes.

B. SI-406 B, Loop 1 Shutdown Cooling Suction Header Relief, lifts and sticks open.

C. LPSI Pump B amps and flow indication become erratic.

D. CCW Pump B trips and alarm "LPSI PUMP B BRNG WATER FLOW LO" annunciates.

ANSWER

A. SI-407 B, Loop 1 Suction Isol Downstream Outside Valve, closes.

COMMENTS

Examination Question Number 74

Description:	Proper communications during the EOPs			**Question ID:**	5738
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OI-038-000	0	0	9/15/98	ADM, PPE

NRC KA Number	RO	SRO	Training Material	Objective
2-4-15	3.0	3.5	W-3-LP-OPS-PPE01	4, 5

QUESTION

The following plant conditions exist:

RCS Pressure is 1100 psia and steady
Pressurizer level is 15% and dropping
S/G #1 level is 20% NR and steady
S/G #1 pressure is 650 psia and dropping
Containment pressure is 27 psia and rising
Tcet is 495 °F and dropping
S/G #2 level is 10% WR and dropping
S/G #2 pressure is 280 psia and dropping

Standard Post Trip Actions of OP-902-000 are completed and OP-902-004, Excess Steam Demand Recovery Procedure is being implemented. Per OI-038-000, Emergency Operating Procedures Operating Expectations, which of the following is an appropriate report to give to the CRS for these conditions?

A. "Emergency Diesel Generator A trouble".
B. "RWSP level Hi/Lo, level 86% and dropping".
C. "Boric Acid Makeup Tank A level Hi/Lo, level 75% and dropping".
D. "Diverse EFW Active/Trouble".

ANSWER

D. "Diverse EFW Active/Trouble".

COMMENTS

Examination Question Number 75

Description:	Determining if Inventory is being met			**Question ID:**	3166
Tier:	3	**Group:**	N/A	**Cognitive Level:**	Comp/Analysis
Source:	Modified Bank (used)	**History:**	RO class 5-18-99 (previous version)		

Reference	Revision	Change	Date	Plant Systems:
OP-902-005	10	0	9/9/98	PPE

NRC KA Number	RO	SRO	Training Material	Objective
2-4-21	3.7	4.3	W-3-LP-OPS-PPE05	1

QUESTION

Plant conditions following a reactor trip due to a loss of offsite power are:

- Pressurizer level - 20% and stable
- Pressurizer pressure - 2000 psia and stable
- Representative CET temperature - 620°F and stable
- RVLMS indicates no levels voided

The RCS inventory safety function is:

A. Being maintained because RVLMS indicates plenum level > 80%.
B. Being maintained because Pressurizer level is ≥ 7% and stable.
C. NOT being maintained because RCS subcooled margin is < 28°F.
D. NOT being maintained because Pressurizer level is < 33%.

ANSWER

C. NOT being maintained because RCs subcooled margin is less than 28°F.

COMMENTS

Use of steam tables required

Waterford 3 Examination Question
Examination Bank

Examination Question Number 76

Description: Fire agents in the Fuel Handling Building

Question ID: 5732

Tier: 3 **Group:** N/A

Cognitive Level: Memory/Fund

Source: New **History:** N/A

Reference	Revision	Change	Date
FP-001-020	11	0	1/27/98

Plant Systems:

FPP

NRC KA Number	RO	SRO	Training Material	Objective
2-4-26	2.9	3.3	W-3-LP-OPS-FP00	10

QUESTION

Which of the following is the proper fire extinguishing agent to be used in the +46 elevation of the Fuel Handling Building?

A. Water fog
B. Dry Chemical
C. Water solid stream
D. Foam

ANSWER

C. Water solid stream

COMMENTS

Examination Question Number 77

Description:	Bypassing second channel of ESFAS			**Question ID:**	5700
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-903-107	14	2	3/17/99	PPS

NRC KA Number	RO	SRO	Training Material	Objective
3.2-013-K5.02	2.9	3.3	W-3-LP-OPS-PPS00	4

QUESTION

The plant is at 100% power. The Channel A PZR Press Lo trip bistable is bypassed due to the Safety Channel A pressure transmitter failing low. Subsequently, Channel D PPS Functional Test is performed as scheduled. Prior to inserting a low Pressurizer pressure signal into Channel D, the SNPO depresses the trip channel bypass pushbutton for the Channel D PZR Press Lo trip bistable. The SNPO then notices his mistake. What is the result of this action if the SNPO stops at this point?

A. Channel A and D are bypassed and trip logic is 1out of 2.

B. Channel A and D are bypassed and trip logic is 2 out of 2.

C. Channel D does **NOT** bypass, Channel A is removed from bypass, a reactor trip and SIAS occur.

D. Channel D does **NOT** bypass, Channel A is removed from bypass, a reactor trip and SIAS do **NOT** occur.

ANSWER

D

COMMENTS

Examination Question Number 78

Description:	Bistables to be bypassed for a failure of the upper detector of a safety NI channel			**Question ID:**	5701
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
TS 3.3.1				NI

NRC KA Number	RO	SRO	Training Material	Objective
2-1-11	3.0	3.8	W-3-LP-OPS-PPS00	9

QUESTION

The plant is at 100% of rated thermal power. A failure of the Channel A excore nuclear instrument middle detector has occurred. All of the following PPS Channel A trip bistables are required to be bypassed within one hour **EXCEPT**:

A. Linear Power HI

B. Log Power HI

C. DNBR LO

D. LPD HI

ANSWER

B

COMMENTS

Examination Question Number 79

Description:	Effect of a malfunction of the pulse height discriminator on the SU channel NIs			**Question ID:**	5702
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
Sys Desc for NI				NI

NRC KA Number	RO	SRO	Training Material	Objective
3.7-015-K6.02	2.6	2.9	W-3-LP-OPS-ENI00	2

QUESTION

The discriminator volts adjust potentiometer in Startup Channel 1 has failed such that discriminator voltage increases. How does this affect Startup Channel 1 when it is energized?

A. Startup Channel 1 reads lower than actual due to fewer gamma pulses meeting the threshold for counting.

B. Startup Channel 1 reads lower than actual due to fewer neutron pulses meeting the threshold for counting.

C. Startup Channel 1 reads higher than actual due to increased amplification of gamma pulses.

D. Startup Channel 1 reads higher than actual due to increased amplification of neutron pulses.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 80

Description:	Effect of open CET thermocouples on RCS Draindown			**Question ID:**	5703
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-001-003	18	2	8/3/99	INI

NRC KA Number	RO	SRO	Training Material	Objective
3.7-017-A2.01	3.1	3.5	W-3-LP-OPS-REQ13	9

QUESTION

Given the following initial conditions:

- The plant is in MODE 5 preparing for entry into MODE 6.
- An RCS drain down is in progress to install nozzle dams in each steam generator.
- PZR level is currently at 30% Cold Cal.
- Jumpers have been installed on 2 CETs per QSPDS channel to facilitate the drain down all other CETs are disconnected.

The PNPO notes that one of the two dedicated CETs from QSPDS 1 is no longer providing valid readings. Which of the following would be required to remain in compliance with OP-001-003, RCS Drain Down.

A. Ensure the failed CET is covered by an active RCS Perturbation Log form.

B. Stop the RCS drain down prior to lowering RCS level less than 5% Cold Cal PZR level.

C. Stop the RCS drain down prior to lowering RCS level below 18 ft MSL.

D. Restore at least 2 CETs from QSPDS 1 prior to continuing any drain down.

ANSWER

A

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 81

Description:	Definition of a Severe Accident in SAMG space			**Question ID:**	5222
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	Bank	**History:**	Not Used		

Reference	Revision	Change	Date
SAMG-01			

Plant Systems:
SAM

NRC KA Number	RO	SRO
2-1-7	3.7	4.4
2-4-38	2.2	4.0

Training Material	Objective
W-3-LP-OPS-SAM00	1, 2

QUESTION

Which of the following would prompt entry into the SAMGs?
(Assume core exit thermocouples are available.)

a. SGTR with concurrent steamline break on same SG

b. Partial core uncovery with CETs > 1200F

c. Small break LOCA with vessel level at centerline of hot legs

d. Large break LOCA with all critical safety functions satisfied

ANSWER

b. Partial core uncovery with CETs > 1200F

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 82

Description:	Effects of containment pressure on status of CCS			**Question ID:**	5704
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-902-009	0	0	9/9/98	CCS

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.5-022-K4.02	3.1	3.4	W-3-LP-OPS-CCS00	2

QUESTION

Given the following conditions:

- Containment Temperature = 160°F
- Containment Pressure = 17.5 psia
- All Containment Fan Coolers (CFCs)are OPERABLE

Determine the expected status of the Containment Cooling System.

A. 3 of 4 CFCs are running in fast speed and discharging through the safety dampers.

B. 4 of 4 CFCs are running in fast speed and discharging through the ring header.

C. 3 of 4 CFCs are running in slow speed and discharging through the ring header.

D. 4 of 4 CFCs are running in slow speed and discharging through the safety dampers.

ANSWER

D

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 83

Description:	Effect of a prolonged loss of Reactor Cavity Cooling			**Question ID:**	5705
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**
W-3-LP-OPS-CCS00			

Plant Systems:
RCC

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.5-022-K3.01	2.9	3.2	W-3-LP-OPS-CCS00	1
			W-3-LP-OPS-CCS00	5

QUESTION

Describe the concern associated with a prolonged loss of the Reactor Cavity Cooling Fans with the reactor at 100% power.

A. CEA drops occur due to overheating of CEDM coils.

B. Reactor Cavity concrete becomes embrittled due to dryout.

C. Increased thermal stresses occur across the reactor vessel metal.

D. Incore instrument leads become brittle due to overheating.

ANSWER

B

COMMENTS

Examination Question Number 84

Description:	Affect of a condensate pump trip on the FWPTs			**Question ID:**	5719
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
OP-003-003	14	1	6/25/99

Plant Systems:
CD, FWP

NRC KA Number	RO	SRO
3.4-056-A2.04	2.6	2.8

Training Material	Objective
W-3-LP-OPS-CD00	5
W-3-LP-OPS-FWP00	2
W-3-LP-OPS-FW00	3

QUESTION

Given the following initial conditions:

- The plant is at 30% power following a turbine trip and Reactor Power Cutback (RPC) Actuation
- FWPT A and B are in service and supplying the steam generators at a discharge pressure of 1100 psig
- Condensate Pumps A, B, and C are running
- Heater Drain Pumps A, B, and C tripped on low suction pressure during the transient
- Plant loads are being supplied by the Startup transformers

The Startup Transformer feeder breaker to the A1 bus trips on overcurrent followed by a trip of FWPT A and a reactor trip. Determine the cause of the FWPT A trip.

A. Low Suction Pressure trip

B. Low Suction Flow trip

C. Condensate Pump Interlock trip

D. Recirculation Failure trip

ANSWER

C

COMMENTS

Examination Question Number 85

Description:	Determine response of S/G level and FWP to a failed open steam bypass valve			**Question ID:**	5707
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
W-3-LP-OPS-MS00				FW, FWC
W-3-LP-OPS-FWP00				

NRC KA Number	RO	SRO	Training Material	Objective
3.4-059-A1.07	2.5	2.6	W-3-LP-OPS-FWC00	4
			W-3-LP-OPS-SBC00	3
			W-3-LP-OPS-MS00	14

QUESTION

The plant is at 60% power with FWPT A in service when steam bypass valve MS-319A fails to 100% open. Which of the following describes the response of steam generator levels and FWPT A speed?

A. Steam generator levels initially swell and then lower; FWPT A speed rises and steam generator levels return to setpoint.

B. Steam generator levels initially swell and then lower; FWPT A speed rises to maximum program speed and steam generator levels continue to lower.

C. Steam generator levels initially shrink, FWPT A speed rises to maximum program speed, then steam generator levels return to setpoint and FWPT A speed lowers.

D. Steam generator levels initially shrink; FWPT A speed rises to maximum program speed and steam generator levels continue to lower.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 86

Description:	Effect of RPC actuation on SG levels with FW in manual			**Question ID:**	5720
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date
W-3-LP-OPS-FWC00			

Plant Systems:
FW, FWC

NRC KA Number	RO	SRO
3.4-059-K3.03	3.5	3.7.

Training Material	Objective
W-3-LP-OPS-FW00	3
W-3-LP-OPS-FWC00	6
W-3-LP-OPS-FW00	4

QUESTION

The plant is at 100% power. FWCS 1 controllers are in manual due to a failure of one control channel level transmitter. Later a Reactor Power Cutback occurs due to a trip of FWPT B. The SNPO is focused on a problem with the turbine runback circuit. What is the effect on the FW system and the steam generators if no operator action occurs?

A. SG 1 level rises, FWCS 1 goes into high level override (HLO), and then SG 1 level will cycle around the HLO setpoint.

B. SG 1 level rises to the reactor trip setpoint, FWCS 1 goes into reactor trip override (RTO), SG level returns to normal values.

C. SG 1 level rises to the reactor trip setpoint and then continues to rise until the transient is terminated by a closure of the MFIV.

D. SG 1 level rises to the reactor trip setpoint, FWCS 1 goes into high level override (HLO), SG 1 level will cycle around the HLO setpoint.

ANSWER

C

COMMENTS

Examination Question Number 87

Description:	EFW valve power supplies			**Question ID:**	5706
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-009-003	11	2	4/21/99	EFW

NRC KA Number	RO	SRO	Training Material	Objective
3.4-061-K2.01	3.2*	3.3	W-3-LP-OPS-EFW00	3

QUESTION

The plant is performing a transfer of the AB safety busses from the A train to the B train. Which of the following EFW pump AB valves will lose power during the transfer?

A. MS-401A, EFW Pump AB MS Supply Isolation A

B. MS-401B, EFW Pump AB MS Supply Isolation B

C. MS-416, EFW Pump AB Turbine Stop Valve

D. MS-417, EFW Pump AB Turbine Governor Valve

ANSWER

C

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 88

Description:	Status of EFW FCVs with failed flow instrument.			**Question ID:**	1786
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	Used Bank	**History:**	RO class 12-11-98		

Reference	Revision	Change	Date
Sys Desc for EFW			

Plant Systems:
EFW

NRC KA Number	RO	SRO	Training Material	Objective
3.4-061-A1.01	3.9	4.2	W-3-LP-OPS-EFW00	05
3.4-061-K4.11	2.7	2.9	W-3-LP-OPS-EFW00	06

QUESTION

Given the following plant status, CHOOSE the proper EFW system response. Assume EFAS-1 and EFAS-2 are the only ESFAS Actuations actuated.

S/G #1	S/G #2
39% WR level (decreasing)	53% WR level (decreasing)
EFW Flow transmitter failed high	EFW flow transmitter operating properly
EFW isolation valves - open	EFW isolation valves - open
EFW Primary FCV - Auto	EFW Primary FCV - Auto
EFW Backup FCV- Auto	EFW Backup FCV - Auto

A. S/G 1 Primary FCV opens to preset valve position, and Backup FCV opens to 400 gpm flow value.

B. S/G 2 Primary FCV opens to preset valve position, and Backup FCV opens to 400 gpm flow value.

C. S/G 1 Primary FCV opens to preset valve position, and Backup FCV is closed.

D. S/G 1 Primary FCV opens to preset valve position, and Backup FCV opens to 175 gpm flow value.

ANSWER

C. S/G 1 Primary FCV opens to preset valve position, and Backup FCV is closed.

COMMENTS

Examination Question Number 89

Description:	Actions required to release a Waste Condensate Tank with the LWM rad monitor OOS			**Question ID:**	66 B
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-007-004	14	3	6/12/98	LWM, PRM
TRM 3.3.3				

NRC KA Number	RO	SRO	Training Material	Objective
3.9-068-K6.10	2.5	2.9	W-3-LP-OPS-LWM00	5

QUESTION

The LWM radiation monitor is out of service. All of the following are required to be performed for a release of a Waste Condensate Tank to the Circulating Water system **EXCEPT**:

A. Independent performance and verification of valve lineup by two technically qualified individuals.

B. Independent performance and verification of release rate calculations by two technically qualified individuals.

C. Two independent samples are taken and analyzed from the Waste Condensate Tank to be discharged.

D. Independent performance of release flow rate estimation every four hours during the Waste Condensate Tank release.

ANSWER

D

COMMENTS

Examination Question Number 90

Description:	Actions for accidental discharge of tanks not on the discharge permit			**Question ID:**	5723
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-414	0	0	4/16/93	BD, PRM

NRC KA Number	RO	SRO	Training Material	Objective
3.9-068-A4.03	3.9	3.8	W-3-LP-OPS-PPO40	3

QUESTION

Given the following conditions:

- SDC train A is in service, maintaining RCS at 135°F
- SG 1 is in wet layup conditions after a Steam Generator Tube Rupture
- SG 2 is being discharged to Circ Water through the Blowdown system
- SG 1 will be discharged to Circ Water following completion of SG 2 discharge
- SG 1 discharge permit has not been prepared and samples have not been taken of SG 1 contents

During the release the PNPO notices that SG 1 level is lowering, Blowdown Containment Isolation Valves, BD-102A and BD-103A are open, the Circ Water Discharge rad monitor is in high alarm, and the automatic isolation has not occurred. What action should be taken for these conditions?

A. Close BD-303, BD to CW or Waste Pond Isolation, to secure the release.

B. Close BD-102 A and BD-103 A to secure the release of S/G 1.

C. Verify BD-109 A, S/G 1 Blowdown Flow Control Vlv, is closed and continue release.

D. Sample S/G 1 and release both S/Gs simultaneously.

ANSWER

A

COMMENTS

Examination Question Number 91

Description:	GWM Compressor automatic operation to control Gas Surge tank pressure			**Question ID:**	5728
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-007-003	13	0	3/8/99	GWM

NRC KA Number	RO	SRO	Training Material	Objective
3.9-071-A3.02	2.8	2.8	W-3-LP-OPS-GWM00	5

QUESTION

Given the following conditions:

- Both GWM compressors are in AUTO
- GWM Compressor B is selected as lead compressor
- The PNPO is venting the VCT to the Gas Surge Header
- Gas Surge Tank Pressure = 14 psig
- Waste Gas Compressor B Clng Water Flow Lo alarm is locked in

Which waste gas compressors should be running for the given conditions?

A. Waste Gas Compressor A

B. Waste Gas Compressor B

C. Waste Gas Compressors A and B

D. Neither Waste Gas Compressor should be running

ANSWER

C

COMMENTS

Examination Question Number 92

Description:	Dose calculation during LOCA for equipment malfunction			**Question ID:**	5745
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
W-3-LP-OPS-RAD02				MCD

NRC KA Number	RO	SRO	Training Material	Objective
3.7-072-K5.01	2.7	3.0	W-3-LP-OPS-RAD02	3

QUESTION

HPSI Pump A is out of service for planned maintenance when a loss of coolant accident occurs. ACC-110 B, ACCW Pump B Discharge, fails to automatically open. The Emergency Coordinator in the TSC authorizes accident mitigating exposure limits and sends a volunteer to open the valve manually. There is a hot point source in the –35 wing area that reads 100 rem/hr at 5 feet. ACC-110 B is 10 feet from the spot. What is the maximum time the volunteer can spend at the valve and not exceed the applicable exposure limits?

A. 5 minutes
B. 10 minutes
C. 20 minutes
D. 30 minutes

ANSWER

C. 20 minutes

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 93

Description:	Affect of CVC-140 failing to the bypass position while on SDC purification			**Question ID:**	5725
Tier:	2	**Group:**	3	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
W-3-LP-OPS-REQ13				CVC, SDC

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.4-005-K1.04	2.9	3.1	W-3-LP-OPS-REQ13	1
			W-3-LP-OPS-REQ13	2

QUESTION

CVC-140, Ion Exchanger Bypass fails to the BYPASS position while on SDC purification. What is the effect of this failure?

A. RCS purification consists of filtration only.

B. An RCS drain is initiated to the VCT or BMS.

C. All SDC purification flow is stopped.

D. Loss of SDC purification cooling occurs.

ANSWER

A

COMMENTS

Examination Question Number 94

Description:	Actions for exceeding 2750 psia and reactor critical			**Question ID:**	5727
Tier:	2	**Group:**	3	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
TS 2.1.2				RCS
OP-100-001	16	0	2/19/99	

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.5-007-A2.03	3.6	3.9	W-3-LP-OPS-RCS00	8
			W-3-LP-OPS-RCS00	9

QUESTION

The reactor is at 100% power when a turbine trip occurs and SBCS fails to operate. The PNPO notes that RCS pressure is 2775 psia and no reactor trip breakers are open. After performing a manual reactor trip, what actions outside of OP-902-000 are required?

A. Restore RCS pressure to within limits of the Pressure/Temperature limits curves within 5 minutes.

B. Restore RCS pressure to within limits of the Pressure/Temperature limits curves within 1 hour.

C. Restore RCS pressure to between 2025 and 2275 psia within 2 hours.

D. Restore a pressurizer code safety to operable and lower RCS pressure to < 2500 psia within 15 minutes.

ANSWER

B

COMMENTS

Examination Question Number 95

Description:	Affect of low ccw surge tank level on pump suction & discharge X-connects with AB assignment switch in A			**Question ID:**	5726
Tier:	2	**Group:**	3	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-002-003	12	1	3/25/99	CC

NRC KA Number	RO	SRO	Training Material	Objective
3.8-008-A1.04	3.1	3.2	W-3-LP-OPS-CC00	3

QUESTION

Given the following initial conditions:

- CCW Pump AB replacing CCW Pump A
- AB Assignment Switch in "A" position
- AB Busses are aligned to the "B" train

CCW surge tank drops to 10%. Describe the effect on CCW Pump AB and the pump suction and discharge cross-connect valves.

A. Pump suction and discharge cross-connect valve to Train A close and CCW Pump AB supplies Train B.

B. Pump suction and discharge cross-connect valve to Train B close and CCW Pump AB supplies Train A.

C. All pump suction and discharge cross-connect valves close and CCW Pump AB trips.

D. All pump suction and discharge cross-connect valves remain open but CCW Pump AB trips.

ANSWER

B

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 96

Description:	When to place Hydrogen Recombiners in service			**Question ID:**	5746
Tier:	2	**Group:**	3	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-902-002	8	0	9/9/98	PPE
				HRA

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.5-028-A1.01	3.4	3.8	W-3-LP-OPS-PPE04	11

QUESTION

A LOCA is in progress and OP-902-002, Loss of Coolant Accident Recovery, is being implemented. Which hydrogen concentration listed is the lowest that requires the hydrogen recombiners to be in service?

A. 1%
B. 2%
C. 3%
D. 4%

ANSWER

A. 1%

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 97

Description:	Quick Open of SBC Valves			**Question ID:**	1235
Tier:	2	**Group:**	3	**Cognitive Level:**	Comp/Analysis
Source:	Used Bank	**History:**	RO class 1-13-99		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
W-3-LP-OPS-SBC00				SBC

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.4-041-K4.18	3.4	3.6	W-3-LP-OPS-SBC00	5
3.4-041-K4.17	3.7	3.9		

QUESTION

The following plant conditions exist: The plant is at full power. Reactor Cutback is out of service. Reactor trip on turbine trip is enabled. A large load rejection occurs which results in a reactor trip.

The response of the steam bypass control system immediately following generation of the reactor trip signal will be: (assume all SBCS valves available)

A. all valves will quick open

B. valves 1 through 5 will quick open

C. valves 1 through 3 will quick open

D. no valves will quick open

ANSWER

B. valves 1 through 5 will quick open

COMMENTS

Examination Question Number 98

Description:	Auto Start of ACC pump due to loss of jockey pump			**Question ID:**	5717
Tier:	2	**Group:**	3	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-002-003	12	1	3/25/99	ACC

NRC KA Number	RO	SRO	Training Material	Objective
3.4-076-K2.01	2.7	2.7	W-3-LP-OPS-CC00	3
			W-3-LP-OPS-CC00	4

QUESTION

Which of the following events/conditions will result in an automatic start of an ACC pump?

A. Loss of power to MCC 313A

B. WCT Basin temperature exceeds 95°F

C. Sequencer test switch taken to the TEST position

D. Manual start of an Emergency Diesel Generator

ANSWER

A

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 99

Description:	Conditions requiring implementation of steps in OP-901-511, IA malfunction			**Question ID:**	5716
Tier:	2	**Group:**	3	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	Revision	Change	Date	Plant Systems:
OP-901-511	4	1	7/15/99	IA

NRC KA Number	RO	SRO	Training Material	Objective
2-4-11	3.4	3.6	W-3-LP-OPS-PPO50	3

QUESTION

IA pressure is noted to lower from 110 psig to 90 psig and stabilizes. The leak is located and isolated 30 minutes after the initial drop in air pressure. All of the following actions are required for this condition **EXCEPT**:

A. Verify all IA and SA air compressors are operating properly.

B. Verify SA backup supply to IA Press Cntl Valve, SA-125 is open.

C. Verify IA air Dryer bypass is open.

D. Place Essential Air in service to safety-related valves.

ANSWER

D

COMMENTS

Waterford 3 Examination Question
Examination Bank

Examination Question Number 100

Description:	TS Loss of Containment Integrity			**Question ID:**	3495
Tier:	2	**Group:**	3	**Cognitive Level:**	Memory/Fund
Source:	Used Bank	**History:**	1996 RO NRC exam		

Reference	Revision	Change	Date	Plant Systems:
TS 3.9.4				CB, TS, RF, FHS

NRC KA Number	RO	SRO	Training Material	Objective
3.5-103-K3.03	3.7	4.1	W-3-LP-OPS-CB00	6
			W-3-LP-OPS-REQ04	02

QUESTION

WHICH ONE (1) of the following conditions is not allowed during Core Alterations per Tech Spec 3.9.4?

A. The Equipment Hatch is closed with ONLY 4 bolts.

B. One Emergency Air Lock door is OPEN.

C. Containment Purge is in progress.

D. The seals are removed from an electrical penetration.

ANSWER

D

COMMENTS

ATTACHMENT 3

ANSWER KEY AND POST ADMINISTRATION COMMENTS

Entergy

Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6242

Early C. Ewing III
Director, Nuclear Safety Assurance
Waterford 3

W3F1-99-0162
A4.05
PR

October 14, 1999

Mr. E.W. Merschoff
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Operator Examination Comments

Dear Mr. Merschoff:

During the week of October 4, 1999, an NRC RO Examination was administered to five candidates at Waterford 3. As discussed with the examination team last week, we are hereby submitting comments (attached) on four of the examination questions. The comments are being submitted per guidance in NUREG 1021, ES-402, Section E. This submittal does not contain any commitments.

If you have any questions concerning the above, please contact R.W. Fletcher at (504) 739-6038.

Very truly yours,

Everett P. Pickering Jr. (for E.C. Ewing)

E.C. Ewing
Director,
Nuclear Safety Assurance

ECE/OPP/ssf
Attachment

(w/o Attachment)
cc: C.P. Patel (NRC-NRR), J. Smith, N.S. Reynolds,
NRC, Document Control Desk, NRC Resident Inspectors Office

Examination Question Number 29

Description:	Instrumentation/equipment affected by a loss of AB-DC			**Question ID:**	5676
Tier:	1	**Group:**	2	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-901-313	1	0	4/19/94	DC

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
4.2-A58-AK1.01	2.8	3.1*	W-3-LP-OPS-PPO30	3

QUESTION

The AB DC bus has developed a short resulting in the loss of both AB battery chargers and the AB Battery. Which of the following is directly affected by the loss?

A. FWPT A and B Governor Valves

B. Main Turbine Trip Circuitry

C. FWPT A and B Trip Circuitry

D. CVC-209, Charging Header Isolation Valve

ANSWER

D

Facility Comment:

The power supply for CVC-209 was changed to the AB DC Bus during Refuel 9. The approved procedures that address the power supply for CVC-209, OP-901-313 (Loss of a 125 Volt DC Bus) and OP-002-005 (Chemical and Volume Control System), have not been changed to reflect the new power supply.
In addition, the applicants' class on the off normal procedures was conducted in the summer of 1999 with OP-901-313 not updated. Listed in the off normal are the alarms, indications lost, and subsequent actions required on a loss of the AB DC Bus. None of these list CVC-209 as being affected by the power loss.

For this reason, we recommend dropping question 29.

B. Symptoms (continued)

1. Alarms (continued)

- Loss of 125 Volt DC Bus AB-DC

a. SUPS 3AB TROUBLE (CABINET C, G-8)

b. BATTERY CHGR SAB1 TROUBLE (CABINET C, L-9)

c. BATTERY CHGR SAB2 TROUBLE (CABINET C, M-9)

d. BATTERY SAB TROUBLE (CABINET C, N-9)

e. GEN 86 RELAY TROUBLE (CABINET D, H-7)

f. DISTANCE 86 RELAY TRIP/TROUBLE (CABINET D, K-7)

g. BUS SAB VOLTAGE MONITOR DC LOST (CABINET D, N-2)

h. 4KV BUS SAB BKR TRIP/TROUBLE (CABINET D, T-2)

i. 480V BUS SAB BKR TRIP/TROUBLE (CABINET D, U-2)

j. CHARGING PUMP AB NOT AVAILABLE/NOT ALIGNED (CABINET G, A-5)

k. EFW PUMP AB UNAVAILABLE (CABINET M, A-10)

l. CCW PUMP A/B TRIP/TROUBLE (CABINET M, E-10)

m. HPSI PUMP A/B TRIP/TROUBLE (CABINET M, H-10)

n. EMER FW TURBINE SG1 SHUTOFF VLV TROUBLE/EFAS OVRD (CABINET M, M-3)

o. EMER FW TURBINE SG2 SHUTOFF VLV TROUBLE/EFAS OVRD (CABINET N, M-13)

B. Symptoms (continued)

2. Indications (continued)

- Loss of 125 Volt DC Bus AB-DC

a. The following indicating lights on CP-15 will be extinguished:

- 86G1
- 86G2
- GENERATOR LOCKOUT RELAY

b. The following meter on CP-1 will fail as indicated:

- AB-DC BUS VOLTAGE (DC-IEI-4613) fails to zero
- NEGATIVE PHASE SEQUENCE CURRENT (GEN-EM-2205) fails to zero

c. The following breakers on CP-1 will exhibit loss of indications:

- BUS AB3 TO A3 TIE BREAKER (AB3S2)
- BUS AB3 TO B3 TIE BREAKER (AB3S6)
- BUS AB31 TO A31 TIE BREAKER (AB31S1B)
- BUS AB31 TO B31 TIE BREAKER (AB31S4A)
- SECONDARY COMPUTER FEEDER BREAKER

d. The following valves on CP-8 will exhibit loss of indication:

- EFW PUMP TURBINE GOVERNOR VALVE (CP-8)
- EMERGENCY FEEDWATER PUMP AB TURB STM SUPPLY SG 1 (MS 401A)
- EMERGENCY FEEDWATER PUMP AB TURB STM SUPPLY SG 2 (MS 401B)

e. All FPMs and FPEs on MRCP will exhibit loss of indication except FPE-A and FPE-B.

C. Automatic Actions (continued)

- Loss of 125 Volt DC Bus TGB-DC

1. ALL Station Air AND Instrument Air compressors will fail to load.

2. Loss of Air Side Seal Oil Backup Pump.

3. Loss of Main Turbine DC Emergency Bearing Oil Pumps.

4. Loss of Emergency Lube Oil Pumps for the Main Feed Pump Turbines.

6. Loss of Voltage Regulator control.

7. The following 4KV AND 7KV switchgear are left without control power. ALL remote manual control AND automatic protection of the switchgear AND associated connected components are disabled:

   - 7KV Bus 1A
   - 7KV Bus 1B.
   - 4KV Bus 2A
   - 4KV Bus 2B
   - 4KV Bus 4A
   - 4KV Bus 4B

NOTE 8

480V breakers overcurrent protection is available from Electronic Current Sensing Device (ECS).

8. The following 480V Switchgear are left without control power. ALL remote manual control of the switchgear AND associated connected components are disabled:

   - 480V Bus 21A
   - 480V Bus 21B
   - 480V Bus 22A
   - 480V Bus 22B.
   - 480V Bus 23A
   - 480V Bus 23B

E_3. Loss of 125 Volt DC Bus AB-DC

```
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
*                      CAUTION E3                           *
*                                                           *
*                                                           *
*                                                           *
*  1. Safety related 4KV bus 3AB is left without control    *
*     power.  All remote manual control and automatic       *
*     protection of these load centers and connected        *
*     components are disabled.                              *
*                                                           *
*  2. 480V breakers will have overcurrent protection        *
*     from ECS, but are left without control power.         *
*                                                           *
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
```

1. Verify Automatic Actions (Section C) take place as designed.

2. Investigate the cause of the AB-DC bus loss.

3. EFW Pump AB will be inoperable. Refer to Technical Specification 3.7.1.2.

4. All FPMs and FPEs (except FPE-A and FPE-B) will be de-energized. Refer to UNT-005-013, Fire Protection Program.

5. CCW AB selector switch control power will be lost. Refer to Technical Specification 3.8.1.2.

6. <u>IF</u> the cause of the AB-DC bus loss is found <u>AND</u> corrected, <u>THEN</u> restore AB-DC in accordance with OP-006-003, 125V DC ELECTRICAL DISTRIBUTION SYSTEM OPERATIONS.

E3. Loss of 125 Volt DC Bus AB-DC (continued)

```
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
*                        CAUTION 7                            *
*                                                             *
*     With the battery bank inoperable, restore to service    *
*     within 2 hours or be in Hot Standby within the next     *
*     6 hours and in Cold Shutdown within the following       *
*     30 hours. (T.S. 3.8.2.1).                               *
*                                                             *
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *
```

7. IF bus AB-DC can NOT be reenergized, THEN conduct a controlled shutdown AND cooldown to Mode 5 in accordance with OP-010-001, GENERAL PLANT OPERATIONS, in conjunction with this procedure.

END

Attachment 5: 3AB-DC-S Loads

CIRCUIT NUMBER	SERVICED COMPONENT	CWD NUMBER	COMPONENT FAILURE ON LOSS OF POWER	PLANT EFFECTS
1	Spare	NA	NA	NA
2	Spare	NA	NA	NA
3	Space	NA	NA	NA
4	Space	NA	NA	NA
5	Space	NA	NA	NA
6	Space	NA	NA	NA
7	Space	NA	NA	NA
8	Spare	NA	NA	NA
9	Battery Charger 3AB1-S DC-EBKR-AB-9	2561	Feeder from Battery Charger to Battery.	Loss of charge ability to battery.
10	Spare	NA	NA	NA
11	Space	NA	NA	NA
12	Space	NA	NA	NA
13	SUPS 3AB ID-EBKR-AB-13	2572	Loss of backup power to SUPS 3AB.	Possible loss of SUPS 3AB.
14	Battery Charger 3AB2-S ID-EBKR-AB-14	2561	Feeder from Battery Charger to Battery (DC-EBATAB).	Loss of charge ability to battery.
15	Spare	NA	NA	NA
16	Spare	NA	NA	NA
17	Spare	NA	NA	NA
18	Spare	NA	NA	NA
19	Spare	NA	NA	NA

Attachment 5: 3AB-DC-S Loads (continued)

CIRCUIT NUMBER	SERVICED COMPONENT	CWD NUMBER	COMPONENT FAILURE ON LOSS OF POWER	PLANT EFFECTS
20	Fire Damper Dist. Pnl 3FDAB-DC FP-EBKR-AB-20	3FDAB	Lose indication on all Fire Damper panels.	None
21	EFP Turb Steam Shutoff (MS-401A) MS-EBKR-AB-21	1535	Fails as is and lose indication and power for control switch.	Will not open on EFAS signal.
22	EFP Turb Steam Shutoff (MS-401B) MS-EBKR-AB-22	1536	Fails as is and lose indication and power for control switch.	Will not open on EFAS signal.
23	Aux Relay Pnl #3 Aux Relays 4KV-EBKR-AB-23	2409	Lose undervoltage relays for 3AB3-S and 3AB31-S.	Lose undervoltage protection.
24	Dry Cooling Tower Vibration Alarms CC-EBKR-AB-24	771	Lose Dry Cooling Tower vibration alarms.	Will not get any new alarms and cannot reset any activated alarm.
25	CP17 Synchronizing GEN-EBKR-AB-25	2211	Lose DC power to synchronizing circuit.	Lose synchronizing ability for OCBs.
26	CP15 Gen Lockout Relay 86G1 GEN-EBKR-AB-26	2201	Lose 86G1 protection including trip of Heater Drain Pumps.	Cannot trip Generator manually.
27	Spare	NA	NA	NA
28	CP15 Gen Lockout Relay 86G2 GEN-EBKR-AB-28	2205	Lose 86G2 protection including trip of Heater Drain Pumps. Lose GEN-EM-2205, Negative Phase Sequence Current meter.	Cannot trip Generator manually. GEN-EM-2205 will fail low.

Attachment 5: 3AB-DC-S Loads (continued)

CIRCUIT NUMBER	SERVICED COMPONENT	CWD NUMBER	COMPONENT FAILURE ON LOSS OF POWER	PLANT EFFECTS
29	Fire Protection Distr Pnl PDP 346AB Normal Feeder DC-EBKR-AB-29	9828	Lose normal power supply to Fire Protection panel PDP 346AB.	Lose all automatic fire protection for FPM 1 through 20, FPE 1 through 5 and the MRCP, and the charcoal filter deluge valves if backup power lost.
30	Fire Protection DIstr Pnl PDP 3461AB Normal Feeder DC-EBKR-AB-30	3398	Lose normal power supply to Fire Protection panel PDP 3461AB.	Lose all automatic fire protection for FPM 21 through 30A if backup power supply lost.
31	4.16KV Swgr 3AB3-S 4KV-EBKR-AB-31	2406	Lose control power and indication to 3AB3-S loads.	Will not be able to operate 3AB3-S loads. All breaker operations will have to be performed at the breakers.
32	480V Swgr 3AB31-S SSD-EBKR-AB-32	2410	Lose control power to and indication 3AB31-S loads.	Will not be able to operate 3AB31-S loads. All breaker operations will have to be performed at the breakers.
33	Fire Protection Distr Pnl PDP 346AB Standby Feeder DC-EBKR-AB-33	9828B	Lose backup power supply to Fire Protection panel PDP 346AB.	Lose all automatic fire protection for FPM 1 through 20, FPE 1 through 5 and the MRCP, and the charcoal filter deluge valves if normal power lost.
34	Fire Protection Distr Pnl PDP 3461AB Standby Feeder DC-EBKR-AB-34	3398A	Lose backup power supply to Fire Protection panel PDP 3461AB.	Lose all automatic fire protection for FPM 21 through 30A if normal power supply lost.
35	Spare	NA	NA	NA
36	Spare	NA	NA	NA

Attachment 5: 3AB-DC-S Loads (continued)

CIRCUIT NUMBER	SERVICED COMPONENT	CWD NUMBER	COMPONENT FAILURE ON LOSS OF POWER	PLANT EFFECTS
37	EFW Turb Pump AB MS Supply Governor (MS-417) EFW-EBKR-AB-37	1541	Fails open.	EFW Turbine pump AB may overspeed.
38	Spare	NA	NA	NA
39	Spare	NA	NA	NA
40	4.16KV Swgr Breaker Test Panel 4KV-EBKR-AB-40	4763	Lose test power supply to 3AB3-S.	None
41	CP15 Distance Lockout Relay 86D GEN-EBKR-AB-41	2209	Lose Distance Relay 86D.	Will lose distance relay protection and OCBs will not trip on phase to phase faults and phase to ground faults.
42	CCW Pumps Common Ckt Isolation Pnl Relays CC-EBKR-AB-42	700	Lose CCW selector switch control power and indications on CP-8.	Cannot select CCW pump AB to replace another CCW pump.

11.4 CHEMICAL AND VOLUME CONTROL SYSTEM BREAKER LINEUP

COMP. #	EQUIPMENT DESCRIPTION	LOCATION	REQUIRED POSITION	PERFORMED BY (INITIAL/DATE)	VERIFIED BY (INITIAL/DATE)
CVC-EBKR-014AB-13	VCT VALVES AND BORIC ACID PUMPS INTERLOCKS	RAB +21 COL 11A & J	ON	_____	_____
CVC-EBKR-31A-3	CHARGING PUMP A MOTOR HEATER	RAB +21 COL 10A & H	ON	_____	_____
CVC-EBKR-31A-5C	CHARGING PUMP A	RAB +21 COL 1OA & H	RACKED OUT	_____	_____
CVC-EBKR-31AB-4C	CHARGING PUMP AB	RAB +21 COL 11A & J	RACKED OUT	_____	_____
CVC-EBKR-31AB-6	CHARGING PUMP AB MOTOR HEATER	RAB +21 COL 11A & J	ON	_____	_____
CVC-EBKR-31B-3	CHARGING PUMP B MOTOR HEATER	RAB +21 COL 11A & L	ON	_____	_____
CVC-EBKR-31B-6C	CHARGING PUMP B	RAB +21 COL 11A & L	RACKED OUT	_____	_____
CVC-EBKR-311B-1OM	VOLUME CONTROL TANK OUTLET ISOLATION (CVC-183)	RAB +21 COL 10A & K	ON	_____	_____
CVC-EBKR-312A-10D	CHEMICAL ADDITION METERING PUMP	RAB +21 COL 11A & H	OFF	_____	_____
CVC-EBKR-312AB-3H	RWSP TO CHARGING PUMPS SUCTION ISOL (CVC-507)	RAB +21 COL 1OA & H	ON	_____	_____
CVC-EBKR-45AB-20	CHARGING HEADER ISOL (CVC-209)	RAB +35 COL 11A & J	ON	_____	_____

Examination Question Number 42

Description:	Determine the amount of Boron required to withdraw Group 6 CEAs out			**Question ID:**	5694
Tier:	2	**Group:**	1	**Cognitive Level:**	Comp/Analysis
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-002-005	14	8	2/21/99	CED

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.1-001-A4.05	3.7	3.7	W-3-LP-OPS-CVC00	11

QUESTION

Given the following conditions:

- The plant has been operating at 100% power for 275 days
- Reg Group 6 CEAs are inserted to 120" for ASI control and ASI is moving to the bottom of the core
- Tavg = 573°
- RCS Boron Concentration = 400 ppm
- Core Burnup = 450 EFPD
- Boric Acid Makeup Tank (BAMT) A and B Boron Concentration = 5500 ppm

Determine the number of gallons of Boric Acid that must be added to the RCS from the BAMTs to withdraw Reg Group 6 CEAs to 150".

A. 122 gallons

B. 138 gallons

C. 144 gallons

D. 244 gallons

ANSWER

A

Facility Comment:

Question 42 tests the candidate's ability to calculate the quantity of acid to add to withdraw Regulating Group 6 rods from 120 inches. In this question, small errors in reading the correct graph can result in large differences within the distractor range. This is shown on the following page. Using numbers from the right graph in the proper calculations yield large differences in the quantity of acid required.

For this reason, we recommend deleting question 42.

From 1.4.5.1 Inverse Boron Worth at EOC (data taken from tabular version of Plant Data Book)

570° F = 115.1 PPM/%Δρ 573° F = 115.68 PPM/%Δρ
580° F = 117.03 PPM/%Δρ

From 1.5.4.1 CEA Worth at HFP and EOC (data taken from tabular version of Plant Data Book)

Reg Group 6 at 125 inches = 0.083 % Δρ Reg Group 6 at 120 inches = 0.10 % Δρ
Reg Group 6 at 100 inches = 0.174 % Δρ

$$\frac{115.68 \text{ PPM}}{\% \Delta\rho} \Big| \; 0.10 \% \Delta\rho = 11.57 \text{ PPM}$$

From OP-002-005, attachment 11.6

RCS volume = 58,140 gals BAM Tanks concentration = 5,500 PPM

$$\text{Volume of Boric Acid} = 58{,}140 \text{ gals} * \frac{11.57 \text{ PPM}}{5{,}500 \text{ PPM}}$$

Volume of Boric Acid required = 122.3 gals

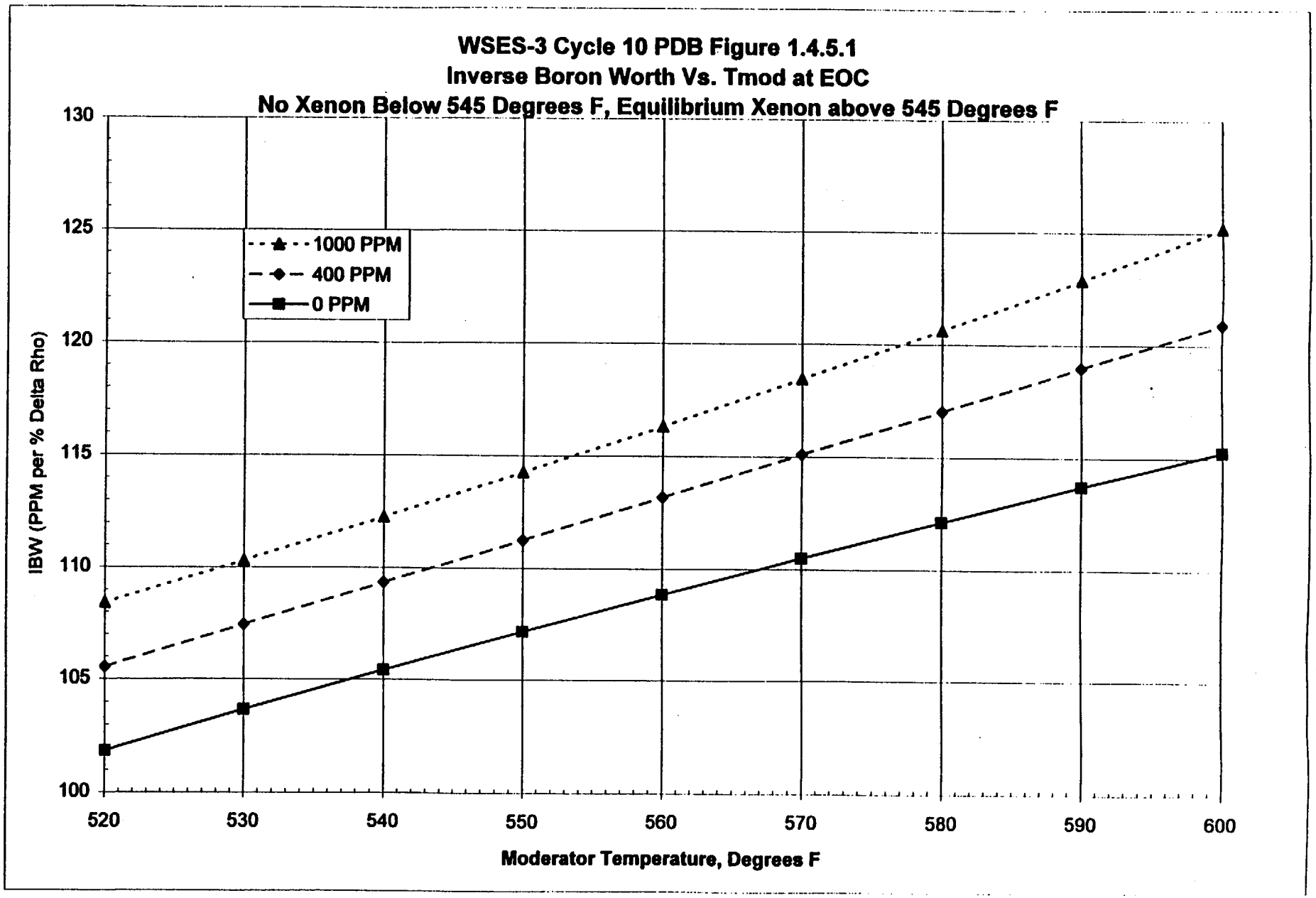
If, however, 0.15 % Δρ was read from the correct graph (1.5.4.1) for CEA worth and used properly within the calculation, the result would have been as follows:

$$\frac{115.68 \text{ PPM}}{\% \Delta\rho} \Big| \; 0.15 \% \Delta\rho = 17.35 \text{ PPM}$$

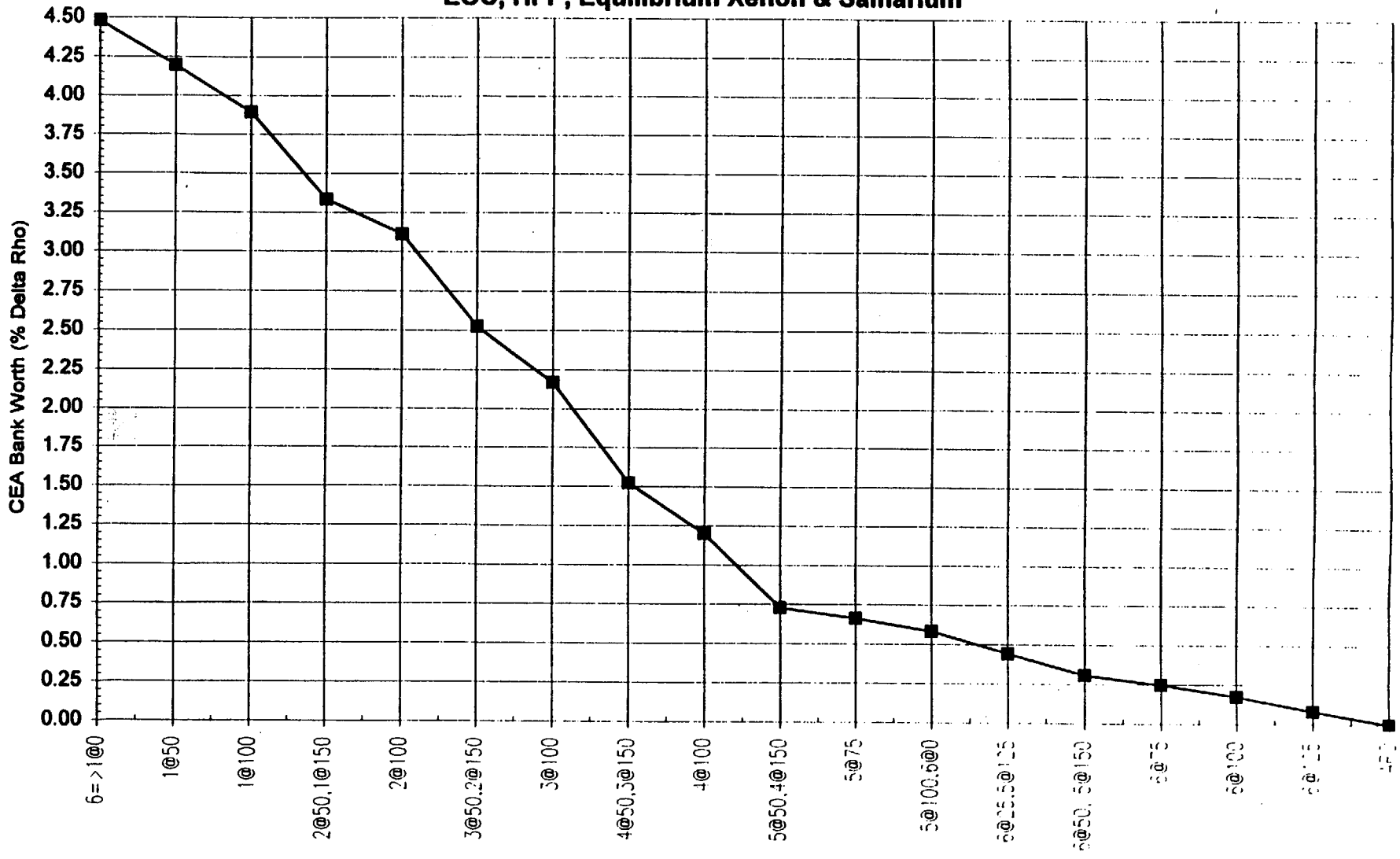
$$\text{Volume of Boric Acid} = 58{,}140 \text{ gals} * \frac{17.35 \text{ PPM}}{5{,}500 \text{ PPM}}$$

Volume of Boric Acid required = 183.4 gals

Using 0.12 % Δρ from graph 1.5.4.1 would require 146.7 gals of boric acid.


WSES-3 Cycle 10 PDB Figure 1.4.5.1
Inverse Boron Worth Vs. Tmod at EOC
No Xenon Below 545 Degrees F, Equilibrium Xenon above 545 Degrees F
1000 PPM
400 PPM
0 PPM
IBW (PPM per % Delta Rho)
130
125
120
115
110
105
100
520
530
540
550
560
570
580
590
600
Moderator Temperature, Degrees F

WSES-3 Cycle 10 PDB Figure 1.5.4.1
CEA Banks 1, 2, 3, 4, 5, 6 Worth Vs. Bank Position
EOC, HFP, Equilibrium Xenon & Samarium



11.6 MAKEUP - BORATE MODE

1. Record the following data:

   BAMT Concentration (CBAMT) _____ PPM

   RCS Boron Concentration (CRCS) _____ PPM

   Final RCS Boron Conc. (CRCSF) _____ PPM

2. Select the appropriate RCS Volume (VRCS) below:

   _____ 0-15% VRCS = 60,532 GAL

   _____ 15-100% VRCS = 58,140 GAL

3. Calculate the volume of Boric Acid (VBA) required:

$$VBA = VRCS \times \frac{CRCSF - CRCS}{CBAMT} = _____ GAL \times \frac{(____ PPM) - (____ PPM)}{(____ PPM)}$$

VBA = _____ GAL

Performed by: _____
Signature & Date

Verified by: _____
Signature & Date

		Examination Question Number	80		
Description:	Effect of open CET thermocouples on RCS Draindown			**Question ID:**	5703
Tier:	2	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	New	**History:**	N/A		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-001-003	18	2	8/3/99	INI

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
3.7-017-A2.01	3.1	3.5	W-3-LP-OPS-REQ13	9

QUESTION

Given the following initial conditions:

- The plant is in MODE 5 preparing for entry into MODE 6.
- An RCS drain down is in progress to install nozzle dams in each steam generator.
- PZR level is currently at 30% Cold Cal.
- Jumpers have been installed on 2 CETs per QSPDS channel to facilitate the drain down all other CETs are disconnected.

The PNPO notes that one of the two dedicated CETs from QSPDS 1 is no longer providing valid readings. Which of the following would be required to remain in compliance with OP-001-003, RCS Drain Down.

A. Ensure the failed CET is covered by an active RCS Perturbation Log form.

B. Stop the RCS drain down prior to lowering RCS level less than 5% Cold Cal PZR level.

C. Stop the RCS drain down prior to lowering RCS level below 18 ft MSL.

D. Restore at least 2 CETs from QSPDS 1 prior to continuing any drain down.

ANSWER

A

Facility Comment:

The requirement to log the loss of the CET is covered in RF-003-001, not OP-001-003. In RF-003-001, it requires the SS to log in the RCS Perturbation Log if continued operation at reduced inventory is required with < 1 CET per QSPDS channel. In this question, there was still 1 operable CET in each QSPDS channel.
In OP-001-003, it list disconnection of a CET for certain refueling operations as an example of what could be an entry in the RCS Perturbation Log. It does not list this as a requirement.

For these reasons, this question contains no correct answer. We recommend deleting this question.

3.0 PRECAUTIONS AND LIMITATIONS (cont'd)

3.2 LIMITATIONS

1. Maintain class "C" cleanliness of Component Cooling Water System in accordance with UNT-007-005, Cleanliness Control.

2. Maintain class "B" cleanliness on sealing surfaces of Cavity Seal Ring, Reactor Vessel Flange and Refueling Cavity, in accordance with UNT-007-005, Cleanliness Control.

3. Cleaning shall be performed with an approved solvent in accordance with UNT-007-003, Control of Consumable Materials.

4. A minimum of two independent Core Exit Thermocouples (CETs) one to each QSPDS, should remain connected to permit core temperature monitoring while operating RCS in a reduced inventory condition.

4.1 **If necessary to operate the RCS in a reduced inventory condition with less than one CET indication per QSPDS channel then the Shift Supervisor shall document this condition in the RCS Perturbation Log in accordance with OP-001-003, Reactor Coolant System Drain Down.**

5. A portable filtration system shall be used to maintain Refuel Cavity water clarity and remove radioactive particles by filtration.

6. Movement of any load in excess of 1500 pounds shall be performed in accordance with UNT-007-008, Control of Heavy Loads.

7. Valve and Breaker manipulations shall be performed by Operations personnel only.

8. All personnel who sign steps or who make log entries within this procedure shall enter their name, signature, and initials on Signature Specimen Sheet found in RF-001-001, Refueling Administration.

3.2.5. The following limitations apply to RCS Perturbations:

3.2.5.1 RCS Perturbation Log and RCS Perturbation Log Index shall be placed in a binder, maintained by the SS, whenever RCS is in Mode 5 or 6.

3.2.5.2 Any activity NOT under direct supervision of Control Room staff that has the potential to affect RCS inventory or heat removal capability shall meet the following requirements:

- Logged on Attachment 11.11, RCS Perturbation Log.
- Evaluated and approved by Shift Supervisor (SS) and Duty Plant Manager (DPM).

3.2.5.3 RCS Perturbations include, but are not limited to, the following:

- Installation and removal of Nozzle Dams
- Freeze seals used as RCS isolations
- Work on any instrumentation that may affect Shutdown Cooling components
- CETs (permanent or temporary) which must be disconnected while performing certain refueling operations
- RCS Drain Down with either RWLIS or RCSLMS unavailable.

3.2.6. The following limitations apply to Containment Closure Impairments:

3.2.6.1 Containment Closure Impairment Log and Containment Closure Impairment Log Index shall be placed in a binder, maintained by the SS, whenever RCS is in Mode 5 or 6.

3.2.6.2 Any activity resulting in a flowpath between inside Containment atmosphere and outside Containment atmosphere, which is NOT provided with an automatic closure on high containment radioactivity, shall meet the following requirements:

- Capable of being isolated within 1 hour.
- Logged on Attachment 11.13, Containment Closure Impairment Log.
- Evaluated and approved by SS and DPM.

3.2.7. All components associated with systems, which have potential to delay or prevent Containment Closure, shall be danger tagged to ensure Containment Integrity.

Examination Question Number 57

Description:	Sequence of events following fault on 3BS load circuit (new SUPS)			**Question ID:**	5528
Tier:	1	**Group:**	1	**Cognitive Level:**	Memory/Fund
Source:	Bank	**History:**	Not Used		

Reference	**Revision**	**Change**	**Date**	**Plant Systems:**
OP-006-005	10	0	3/12/99	ID

NRC KA Number	**RO**	**SRO**	**Training Material**	**Objective**
4.2-A57-AA2.17	3.1	3.4	W-3-LP-OPS-ID00	1
3.6-062-A2.02	2.2	2.6	W-3-LP-OPS-RF00	3

QUESTION

A 100% short to ground on the circuit to CVC-103 has occurred. The power supply is from a circuit on PDP 91B, which is supplied from 3BS vital SUPS. Which of the following sequence of events will occur when the SUPS senses the fault?

A. The SUPS will automatically shut down.
B. The transfer switch will automatically transfer to the alternate supply, the circuit breaker for supply to CVC-103 will open on overload, and the transfer switch will return to the normal source.
C. The transfer switch will automatically transfer to the alternate supply, the circuit breaker for supply to CVC-103 will open on overload. The SUPS can then be returned to normal source by the operator using the "inverter to normal" pushbutton.
D. The circuit breaker supplying CVC-103 will open and the SUPS will remain in service.

ANSWER

B

Facility Comment:

SUPS B is a single phase, 60 hertz, ungrounded power supply. This question describes a 100% short to ground. On an ungrounded power supply, no breakers will trip nor will any transfers occur on a short to ground. Had the stem not specified that the short was to ground, then the question would have been correct.

For this reason, we recommend deleting this question.

MAJOR COMPONENTS

The components of the ID system are the SUPS. All nine of the SUPS are located in the electrical equipment rooms of the Auxiliary Building EL +21. Each type of SUPS will be discussed separately.

VITAL SUPS (SUPS A and B)

The Vital SUPS power vital plant loads which are not powered by the Instrument SUPS. Fig. 02 illustrates a Vital SUPS. The Vital SUPS receive their normal and bypass power from 480 VAC safety related MCCs. Should the normal power supply fail, the SUPS will automatically receive power from a 125 VDC battery. The Vital SUPS are rated for 10 KVA. Table 1 lists the power supplies for the Vital SUPS and the external PDPs to which the SUPS provide power.

The Vital SUPS have Manual Bypass Switches which allow the bypass AC power source to energize the PDPs during SUPS maintenance or failure.

The output of the Vital SUPS is single phase, 60 Hz, ungrounded 120 VAC. The output is ungrounded to provide additional isolation of the equipment powered by these SUPS. Fig. 03 depicts the instrumentation and indication panel for the Vital SUPS. Fig. 04 depicts the control panel for the Vital SUPS. The purpose and function of the controls and indicators follows.

Vital SUPS Instrumentation

Each Vital SUPS is provided with an AC voltmeter which can be used to monitor either the AC output voltage of the inverter or the AC voltage of the bypass supply. A selector switch below the voltmeter is used to select which AC source is to be monitored. The meter has a range of 0-150 VAC.

A DC voltmeter is provided with each Vital SUPS to monitor either the SUPS rectifier voltage or the battery voltage. A selector switch below the voltmeter is used to select the DC source to be monitored. The meter range is 0-150 VDC.

Each SUPS is provided with an AC ammeter to monitor the load current. The ammeter range is 0-150 AC amps.