U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS: 50-272/94-08
50-311/94-08

LICENSEE: Public Service Electric and Gas Company (PSE&G)
Hancocks Bridge, New Jersey

FACILITY: Salem Nuclear Generating Station, Units 1 & 2

DATES: May 23 to June 24, 1994

INSPECTORS: James S. Stewart, Operations Engineer
R. Pelton, Training Specialist, NRR

EXAMINERS: M. Parrish, INEL
S. Johnson, INEL

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INSPECTOR: James S. Stewart 7-15-94
James S. Stewart, Operations Engineer Date
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APPROVED BY: James A. Pellet for 7/21/94
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EXECUTIVE SUMMARY

Background

From May 23 to June 24, 1994, the NRC performed an inspection of the Salem Nuclear Generating Station, Units 1 and 2 licensed operator requalification training program. Also, initial licensed operator examinations were administered to seven employees who had applied for licenses to operate these two units.

Operations

Licensed Operator Requalification: The inspectors reviewed the licensed operator requalification program and the conduct of the annual operating examinations and concluded that these activities were effective in ensuring the continued safe operation of the two units. The requalification program was determined to be based on the systematic approach to training method. Requalification examinations were found to be effective in discriminating safe from unsafe operations, and no serious deficiencies were identified. Two operators, who had failed the written portion of the annual examination, had previously demonstrated consistent weak performance on quarterly quizzes. This shows a weakness in the facility remedial training program. The inspectors determined that excellent evaluations were performed on individual and crew performances following the conduct of their annual operating examinations. The evaluations were thorough and detailed such that reconstruction of operator actions was easily performed for those scenarios not observed by the inspectors. A number of deficiencies in job performance measure administration, identified in an inspection conducted in May 1993, had been adequately corrected. One unresolved item was identified regarding the means and procedure for depressurizing a steam generator during conduct of the emergency operating procedures (UNR 50-311/94-08-01)

Licensed Operator Examinations: All four reactor operators (ROs) and three senior reactor operators (SROs) passed the NRC-administered examinations and were issued licenses. All seven individuals were well prepared for the examination, as demonstrated by their performance during the operating examination and the results of the written examination. Crew communications during the scenario portion of the examination were not as effective as compared to previous examinations; however, it did not deter the crews from achieving safe plant shutdown conditions in all cases observed.

DETAILS

1.0 INSPECTION/EXAMINATION SCOPE AND OBJECTIVES

An announced inspection of Salem operations training and requalification program was conducted from May 23 to June 24, 1994. The scope of the inspection included review and observation of operations training, including administration of the annual operating examination, required by 10 CFR 55.53. The inspection objectives included verification that the requalification program administered to Salem operators adequately evaluated how well the individual operators have mastered training and performance objectives. The inspection included an assessment of the PSE&G effectiveness in ensuring that individuals licensed to operate the Salem facility had satisfied the license requirements of 10 CFR 55.53.

An initial licensed operator examination was also administered by the NRC to seven PSE&G employees who had applied for licenses to operate the Salem facility. The examination was reviewed and validated by the NRC with PSE&G training and operations representatives prior to administration. This review was performed to ensure that the examination was content-valid.

2.0 INSPECTION RESULTS

The inspectors reviewed the annual licensed operator requalification examination, and observed its administration to a number of operating crews and individuals. Overall, all parts of the examination were determined to be appropriate, with good sampling, proper administration, and adequate evaluation.

The Salem requalification training program was determined to be in conformance with the systematic approach to training (SAT) methodology. Learning objectives and training were in conformance with a completed job task analysis, and appropriate feedback systems were in place to evaluate training effectiveness. No program deficiencies were identified.

2.1 Written Examination

No deficiencies were identified in review or in observation of the written classroom and static simulator portions of the 1994 requalification examination. Two requalification exam failures, in this part of the examination, were identified by PSE&G during evaluation of this part of the examination. In both cases, the individuals were removed from licensed duties and placed in a remediation training program. The remediation had not been completed at the end of the inspection.

2.2 Simulator Examination

The performance standards and evaluation of the dynamic simulator examination were determined to be effective in discriminating safe operators. A number of concerns were identified by the inspectors during NRC observation of this portion of the examination:

1. Some supervisory weakness was identified by the facility evaluators for a number of senior operators during conduct of the dynamic simulator examination. Specifically, command and control weakness was identified by the facility when, on a number of occasions, shift supervisors approached the control boards for diagnosis of plant conditions rather than using reactor operators for plant evaluation. One crew-critical task was failed, in one case, when communication breakdown between the senior operator and the balance of plant operator allowed auxiliary feedwater flow to fall below an emergency operating procedure prescribed minimum. This condition was prohibited by emergency operating procedures (EOPs). The facility appropriately failed the individuals and crew involved in the missed critical task.

   The inspectors reviewed the remediation program for the failed crew, observed the retake examination, and identified no deficiencies. The remediation and retesting of the failed crew was appropriate to the failure, was well planned, and properly administered. The failed crew had been removed from licensed duties while the remediation plan was prepared and administered. The crew was retested and all tasks were completed satisfactorily.

2. A technical adequacy and procedure compliance issue was raised by the inspectors when, during the conduct of a simulator scenario, the operating crew shut all steam generator main steam isolation valves (MSIV) without specific direction in the EOPs. Specifically, 2-FRHS-1, Step 26, directs that an intact steam generator be depressurized by opening the associated atmospheric steam dump valve. When completing this step during a scenario, the associated crew isolated all main steam isolation valves to limit the depressurization to one steam generator, in apparent conformance with unwritten facility expectations. The inspectors questioned why condenser steam dumps were not used for the depressurization since the condenser steam dump valves and the main condenser were available in the particular scenario and, further, why the procedure directed opening a single steam generator atmospheric valve without first directing the closing of the associated MSIV. The specific safety issues raised by the inspectors included the potential unmonitored release that could occur if atmospheric steam dumps were used instead of available main condenser steam dumps, the unnecessary isolation of main steam lines and resulting operational limitations including loss of main feedwater pump operation, and procedural compliance involved with isolation of MSIVs without specific direction in the EOPs. Final determination by the NRC of the acceptability of the operator actions in completing 2-FRHS-1, Step 26, which included closing all MSIVs when depressurizing a single steam generator is considered unresolved, pending facility review of procedural intent and adequacy. **(UNR 50-311/94-08-01)**

3. The inspectors identified that a number of facility-selected crew-critical tasks did not meet all of the criteria specified by Examiner Standard ES-604. Specifically, on a number of occasions, the critical task to either "trip the reactor" or "initiate safety injection" involved a manual action that, if not taken, would have resulted in the same

action occurring automatically. Automatic actuation as a backup to a crew critical task is contrary to the guidelines of ES-604. The facility agreed to review selection of crew-critical tasks in future examinations to ensure that the criteria of ES-604 were considered.

4. One simulator fidelity issue was identified during the conduct of the dynamic simulator examination, when wide-range steam generator level did not track during a steam generator level transient. The failure had been previously identified by the facility, a deficiency report had been written, but troubleshooting had not resolved the discrepancy at the time of the requalification examination. Because tracking of wide range level is only occasionally erratic, no compensatory measures were instituted by the facility during the examination. The actions taken by the facility to identify and correct the fidelity problem, although incomplete, were considered appropriate by the inspectors.

2.3 Job Performance Measures

The inspectors reviewed selected job performance measures, observed conduct of the JPM evaluations, and reviewed completed evaluations of performance. Three of five senior reactor operators did not successfully complete a common JPM that required classification of an emergency event. The inspectors reviewed and observed remedial actions by the facility for the operators, and considered the remediation appropriate. Although required by job description to classify events, none of the operators held the senior nuclear shift supervisor position and, therefore, would not have been responsible at Salem for event classification of reactor transients.

Weaknesses in job performance measures administration, identified in a previous requalification inspection, had been corrected, and no deficiencies were identified. Some of the weaknesses specifically corrected included consistency in administration and documentation by the evaluators, use of correct procedures and procedure revisions by the operators, and appropriate preparation for examination administration by the evaluators. None of these previously-identified deficiencies were observed during this inspection.

2.4 Operator Interviews

Interviews were conducted with one operating crew of licensed individuals, as well as a number of training instructors and supervisors. The results of these interviews are summarized below:

1. A number of operators stated that industry events training could be improved. Currently, there are many events that are provided as required reading where there appears to be no direct applicability to Salem operations. It was noted that recent quizzes had included test items from the required reading material.

2. A number of operators identified areas where training could be enhanced, including additional simulator training on balance of plant abnormal procedures, emergency plan implementation in the control room, and the SROs could use more board time to become more familiar with the actions/operations of the control operators. To enhance communication between operators and training, the facility stated an intention to assign one training individual to each crew as a training liaison. This liaison program was under development at the time of the inspection, and the plans had not been finalized.

2.5 Review of remedial training

The inspectors reviewed facility remediation used in response to deficiencies in performance by individual operators, and found the overall program to be appropriate, but not always effective.

Training conducted in response to an operational event was found by the inspectors to be appropriate. On April 7, 1994, a serious grass intrusion from the Delaware River into the Salem 1 circulating water intake structure caused a degraded main condenser vacuum. A number of personnel errors by reactor operators, coupled with a number of equipment deficiencies resulted in a reactor trip, safety injection, and an alert declaration by PSE&G management. The licensed operator errors included poor communication between operators during a rapid downpower, poor command and control by control room supervision, and a number of equipment control errors by the control operators. A full description of the event can be found in NRC Inspection Report 50-272/94-80. Because of the performance deficiencies, all licensed operators at Salem were remediated following the event as part of the plant restart program conducted by the facility. The training included a simulator demonstration of the course of the event, a discussion of the weak operations performance, a review of procedure revisions made subsequent to the event, and a discussion of management expectations.

Licensed operator performance on training cycle quizzes was reviewed by the inspectors, and remedial action for poor quiz performance was not always effective in ensuring satisfactory performance on the annual requalification examination. Specifically, two reactor operators who had failed the written portion of the annual requalification examination had demonstrated consistent poor performance on quarterly quizzes. The inspectors noted that these operators had been involved in the April 7 event. The facility stated, and the inspectors acknowledged, that the training quizzes were not prepared using the same quality assurance checks used in the annual operating test; however, consistent weak performance demonstrated some level of retention or comprehension weakness on the part of operators. The inspectors identified a number of other licensed operators who had demonstrated consistent weak performance on quizzes but had satisfactorily completed the annual examination. Facility management acknowledged that consistent weak performance on training quizzes could be indicative of inadequate operator readiness and stated an intention to review remediation activities for poor quiz performance.

2.6 Management Oversight and Involvement

Station management involvement in the requalification training program was evident. The station operations management routinely observed operator conduct during evaluated simulator sessions each training cycle. Also, operations management evaluated crew performance in the simulator portion of the annual operating test. Weak performance of licensed operators on training quizzes had been routinely reported to operations management by the training department; however, no feedback to either operators or training had been identified by the inspectors. During management interviews, both training and operations managers stated an intention to develop better coordination of training and operator performance.

3.0 INITIAL EXAMINATION RESULTS

3.1 Summary of Results

	SRO Pass/Fail	RO Pass/Fail
Written	3/0	4/0
Simulator	3/0	4/0
Walkthrough	3/0	4/0
Overall	3/0	4/0

3.2 Preexamination Review

Prior to the administration of the RO and SRO written examinations, four Salem training and operations department staff members, under security agreement, reviewed the examinations at the Salem training facility. The review, conducted on June 8, 1994, was performed to ensure that the examinations to be administered during the week of June 20, 1994, were content-valid and performance-based.

Also, all simulator scenarios and JPMs to be utilized during the examination were validated by the NRC on the Salem simulator during June 8-10, 1994. Verification was performed with the assistance of a simulator operator and training staff personnel who were also under the security agreement.

3.3 Examination Overview

3.3.1 Written Examination

The written examinations were administered on June 20, 1994, at the Salem Nuclear Training Center. The examinations, both reactor operator and senior reactor operator, were developed in accordance with the guidelines of 10 CFR 55.41, 55.43, and NUREG-1022, "Examiners Handbook for Developing Operator Licensing Written Examinations." Each examination consisted of 100 questions written in multiple choice format. Seventy-five (75) of the questions were common to both examinations. The examinations and answer keys are enclosed as Attachments 1 and 2 to this report. The training department was provided a copy of the as-administered written examinations immediately after their administration to provide comments for any validity issues with the examination questions. No formal comments were provided to the NRC; however, the facility did note that one question, common to both the RO and SRO exams, did have two correct answers. Upon further review of the question together with backup documentation provided by the facility, the NRC agreed that the question did indeed have two possible correct answers. The answer keys were, therefore, changed accordingly.

3.3.2 Weaknesses Identified During the Post-Examination Review

During a review of the graded written examinations, the following generic areas of weakness were identified. A weakness is considered generic if one-half or more of the personnel taking the examination missed the same question. Question numbers are provided in parentheses. This information is being provided to assist in upgrading initial and requalification training programs. No response to the below listed items is needed.

- Response of the CVCS if makeup mode selector switch is in auto, and VCT level decreases to M/U setpoint (RO-#32; SRO-#31); and

- Conditions that will result in the automatic start of both diesel-driven fire pumps (RO-#43; SRO-#40).

3.3.3 Operating Test

The operating tests were administered from June 21-24, 1994. The operating tests consisted of at least two - and no more than three - dynamic simulator scenarios; 10 JPMs for each of the four ROs; and three instant SROs. Two oral questions were asked with each JPM. All candidates were examined concerning administrative requirements of the Salem Nuclear Generating Station, in addition to the scenarios and JPMs. The examiners noted that many candidates experienced some difficulty in locating the answers to several administrative questions. This difficulty was evidenced by not only the amount of time that it took, but also the number of procedures that were reviewed before finally arriving at the correct answer.

3.3.4 Dynamic Simulator Examination

The candidates were divided into crews for the dynamic simulator examination. Each crew was made up of two RO (board and desk) positions and one SRO position. All crews were successful in completing the scenarios under which they were examined. Communications varied from one crew to another, and were determined to be not as effective or concise as witnessed during the previous examination administered in December 1993 by the same examiner.

3.3.5 Job Performance Measures (JPMs)

The candidates generally exhibited good performance on the seven JPMs administered, either on the simulator or in the control room. The examiners noted, however, that instant SROs without any prior plant experience, i.e., auxiliary operator, equipment operator, etc., exhibited some difficulty in locating various valves and switches during the performance of the in-plant JPMs. Although candidates were generally able to eventually locate the necessary valves and switches essential to satisfactorily completing a particular JPM, it was often done with much difficulty and delay. All candidates demonstrated a thorough working knowledge of personal and radiological safety practices.

For those few JPMs that were not completed successfully, it was not so much a knowledge deficiency as it was a failure on the part of the candidate to fully follow procedures as written.

4.0 EXIT MEETING

Separate exit meetings were held at the conclusion of the inspection and the examination with PSE&G representatives on May 27 and June 24, 1994. PSE&G representatives acknowledged the NRC findings and conclusions. A listing of exit meeting attendees is provided below:

Public Service Electric and Gas, Salem 1 and 2

*L. Catalfumo	Operations Manager
*B. Gott	Principal Training Supervisor, Hope Creek
●J. Lloyd	Principal Training Supervisor, Salem
●G. Mecchi	Nuclear Training Manager
●P. O'Donnell	Operations Engineer

* Denotes those attending the May 27, 1994, exit meeting.
● Denotes those attending the May 27, 1994, and June 24, 1994, exit meetings.

Attachments:
1. RO Exam and Master Examination Answer Key
2. SRO Exam and Master Examination Answer Key
3. Simulation Facility Report

ATTACHMENT 1

RO EXAM AND MASTER EXAMINATION ANSWER KEY

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 1

CANDIDATE'S NAME: _____

FACILITY: Salem 1 & 2

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 94/06/20

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%
100.00	_____	___ % TOTALS

FINAL GRADE

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

_____
Candidate's Signature

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001	a	b	c	d	___
002	a	b	c	d	___
003	a	b	c	d	___
004	a	b	c	d	___
005	a	b	c	d	___
006	a	b	c	d	___
007	a	b	c	d	___
008	a	b	c	d	___
009	a	b	c	d	___
010	a	b	c	d	___
011	a	b	c	d	___
012	a	b	c	d	___
013	a	b	c	d	___
014	a	b	c	d	___
015	a	b	c	d	___
016	a	b	c	d	___
017	a	b	c	d	___
018	a	b	c	d	___
019	a	b	c	d	___
020	a	b	c	d	___
021	a	b	c	d	___
022	a	b	c	d	___
023	a	b	c	d	___
024	a	b	c	d	___
025	a	b	c	d	___
026	a	b	c	d	___
027	a	b	c	d	___
028	a	b	c	d	___
029	a	b	c	d	___
030	a	b	c	d	___
031	a	b	c	d	___
032	a	b	c	d	___
033	a	b	c	d	___
034	a	b	c	d	___
035	a	b	c	d	___
036	a	b	c	d	___
037	a	b	c	d	___
038	a	b	c	d	___
039	a	b	c	d	___
040	a	b	c	d	___
041	a	b	c	d	___
042	a	b	c	d	___
043	a	b	c	d	___
044	a	b	c	d	___
045	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

046	a	b	c	d	___	068	a	b	c	d	___
MULTIPLE CHOICE						069	a	b	c	d	___
047	a	b	c	d	___	070	a	b	c	d	___
048	a	b	c	d	___	071	a	b	c	d	___
049	a	b	c	d	___	072	a	b	c	d	___
050	a	b	c	d	___	073	a	b	c	d	___
051	a	b	c	d	___	074	a	b	c	d	___
052	a	b	c	d	___	075	a	b	c	d	___
053	a	b	c	d	___	076	a	b	c	d	___
054	a	b	c	d	___	077	a	b	c	d	___
055	a	b	c	d	___	078	a	b	c	d	___
056	a	b	c	d	___	079	a	b	c	d	___
057	a	b	c	d	___	080	a	b	c	d	___
058	a	b	c	d	___	081	a	b	c	d	___
059	a	b	c	d	___	082	a	b	c	d	___
060	a	b	c	d	___	083	a	b	c	d	___
061	a	b	c	d	___	084	a	b	c	d	___
062	a	b	c	d	___	085	a	b	c	d	___
063	a	b	c	d	___	086	a	b	c	d	___
064	a	b	c	d	___	087	a	b	c	d	___
065	a	b	c	d	___	088	a	b	c	d	___
066	a	b	c	d	___	089	a	b	c	d	___
067	a	b	c	d	___	090	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

091 a b c d ___

MULTIPLE CHOICE

092 a b c d ___

093 a b c d ___

094 a b c d ___

095 a b c d ___

096 a b c d ___

097 a b c d ___

098 a b c d ___

099 a b c d ___

100 a b c d ___

(********** END OF EXAMINATION **********)

QUESTION: 001 (1.00)

Which of the following individuals is responsible for daily visual accountability of the security keys on the Unit 1 Primary Operator key ring?

a. Primary Operator

b. Shift Clerk

c. Nuclear Control Operator

d. Control Room Nuclear Shift Supervisor

ANSWER: 001 (1.00)

c.

REFERENCE:

Key Control, AD-37, Rev. 11, page 4.
Facility Learning Objective not identified.

[3.1/3.4*]

194001K105 ..(KA's)

QUESTION: 002 (1.00)

An Abnormal Operating Procedure is being performed that contains the following step: "CLOSE MOV 1SW26."

Select when the next step in the procedure can be performed.

a. After an operator has been directed to perform the action and the operator acknowledges the order.

b. After verifying valve is stroking in the closed direction.

c. Anytime because the procedure does not prohibit sequential performance of the steps.

d. After the valve is closed.

ANSWER: 002 (1.00)

d.

REFERENCE:

Emergency/Abnormal Operating Procedures Program, AD44, rev. 3, page 16. Use and Control of Procedures and Directives, AR1B00, Learning Objective 4.

[4.1*/3.9]

194001A102 ..(KA's)

QUESTION: 003 (1.00)

A locked throttle valve is include on a valve lineup that is to be independently verified.

Select the required action for performing the lineup and independent verification for this valve.

a. The initial verifier and independent verifier will simultaneously verify the lock installed and that the valve appears to be in the correct position.

b. The initial verifier will unlock the valve, fully open the valve, reclose it the required number of turns, then relock the valve. The independent verifier will verify the valve correctly locked.

c. The initial verifier will unlock the valve, fully close the valve, reopen it the required number of turns, then relock the valve. The independent verifier will observe the actions of the initial verifier.

d. The initial verifier will unlock the valve, fully close the valve, reopen it the required number of turns, then relock the valve. The independent verifier will use other methods to verify position where possible.

ANSWER: 003 (1.00)

c.

REFERENCE:

System Alignments, SC.OP-DD.ZZ-OD7, Rev 0, page 3 and 4.
Station Operating Practices, AR1A00-09, Learning Objective 4.

[3.6/3.7]

194001K101 ..(KA's)

QUESTION: 004 (1.00)

As second verifier, you identify a valve that is closed, however, its required to be in the open position.

Select the required action.

a. Open the valve. Note on the valve lineup that the valve was found in the open position.

b. Contact the initial verifier. After the initial verifier positions the valve perform the independent verification.

c. Open the valve. Notify the Nuclear Shift Supervisor.

d. Immediately contact the Nuclear Shift Supervisor for resolution.

ANSWER: 004 (1.00)

d.

REFERENCE:

System Alignments, SC.OP-DD.ZZ-OD7, Rev 0, page 2.
Station Operating Practices, AR1A00-09, Learning Objective 4.

[3.6/3.7]

194001K101 ..(KA's)

QUESTION: 005 (1.00)

A Tagging Request and Inquiry System (TRIS) work sheet is being used to tag equipment. The operator performing the tagging recognizes that the valve number is for a different valve than the description of the valve.

Select the required action.

a. Pen and ink change the worksheet and have the Control Room SNSS or NSS initial the change.

b. Pen and ink change the worksheet and have the Work Control Center (WCC) NSS initial the change.

c. Notify the NSS and obtain permission to perform the tagging on the component.

d. Return the worksheet to the WCC and obtain a corrected worksheet.

ANSWER: 005 (1.00)

d.

REFERENCE:

Safety Tagging Program, NC.NA-AP.ZZ-0015, Rev 1, Page 13.
Station Safety Tagging, AR30000-04, Learning Objective 5.

[3.7/4.1]

194001K102 ..(KA's)

QUESTION: 006 (1.00)

What is the administrative annual limit on Total Effective Dose Equivalent (TEDE) for category 2 workers at Salem?

a. 0.5 rem

b. 1.0 rem

c. 2.0 rem

d. 3.0 rem

SWER: 006 (1.00)

c.

REFERENCE:

Radiation Protection Program, NC.NA-AP.ZZ-0024, Rev 3, Attachment 2.
Radiological Control Program, RPP000-05, Learning Objective 2.

[2.8/3.4]

194001K103 ..(KA's)

QUESTION: 007 (1.00)

Which of the following would be the maximum dose rate that would NOT require a door to be locked?

a. 99 mrem/hr

b. 499 mrem/hr

c. 999 mrem/hr

d. 1999 mrem/hr

ANSWER: 007 (1.00)

c.

REFERENCE:

Radiation Protection Program, NC.NA-AP.ZZ-0024, Rev 3, Attachment 2.
Radiological Control Program, RPP000-05, Learning Objective 3.

[2.8/3.4]

194001K103 ..(KA's)

QUESTION: 008 (1.00)

An instrument technician is required to enter a cabinet that is posted "Trip Hazard."

Which of the following identifies the restrictions associated with entry into this cabinet?

a. The posting is only to warn the technician to exercise extra care. No specific permission for entry is required.

b. The technician is authorized entry verbally by the Nuclear Control Operator (NCO).

c. The technician is authorized entry verbally by the Nuclear Shift Supervisor.

d. The technician is authorized entry into the cabinet by the signed authorization of a work order.

ANSWER: 008 (1.00)

d./C

REFERENCE:

Station Operating Procedures, NC.NA-AP.ZZ-0005, Rev. 5, page 11.
Facility Learning Objective Not Identified.

[3.1/4.1*]

194001A112 ..(KA's)

QUESTION: 009 (1.00)

A reactor operator trainee is assigned to your shift. A reactor startup is to be performed.

Who is required to grant permission for the trainee to withdraw the control rods to perform the startup?

a. The trainee is granted permission by being enrolled in the licensing class.

b. The NSS must grant permission for the trainee to perform the startup.

c. The SNSS must grant permission and obtain verbal concurrence from the Operations Manager.

d. The Operations Manager must issue written authorization.

ANSWER: 009 (1.00)

d.

REFERENCE:

Station Operating Procedures, NC.NA-AP.ZZ-0005, Rev. 5, page 28.
Station Operating Practices, AR1A00-09, Learning Objective 3.

[2.8/4.1*]

194001A111 ..(KA's)

QUESTION: 010 (1.00)

What is the length of time that a "Working Copy" of a procedure is valid?

a. 24 hours.

b. 48 hours.

c. 7 days.

d. 14 days.

ANSWER: 010 (1.00)

d.

REFERENCE:

Document Control Program, NC.NA-AP.ZZ-0003, Rev 2, page 9.
Use and Control of Procedures and Directives, AR1B00-07, Learning Objective 5.

[3.3/3.4]

194001A101 ..(KA's)

QUESTION: 011 (1.00)

The plant was operating at 45% when the feedwater regulating valve (BF-19) for 12 Steam Generator was placed in manual due to oscillations in the controller.

What is the requirement for operation with the valve in manual?

a. The Board NCO shall remain in the vicinity of the controller and only perform actions that will not interfere with monitoring the steam generator.

b. The Board NCO is required to announce that a valve has been placed in manual so that all control room personnel can provide additional monitoring of steam generator level.

c. An NCO, other than the Board NCO, is required to be assigned to monitor steam generator level. This individual can perform other tasks that do not interfere with monitoring of steam generator level.

d. An NCO, other than the Board NCO, is required to be continuously stationed at the BF-19 controller. The NCO can have no other responsibilities.

ANSWER: 011 (1.00)

d.

.EFERENCE:

Salem Operations Standing Orders, item Q, page 13.
Facility Learning Objective not Identified.

[2.7/3.9*]

194001A109 ..(KA's)

QUESTION: 012 (1.00)

Which of the following positions would be responsible for gathering and transmitting operational data to the Emergency Operating Facility (EOF) during an emergency?

a. Primary Communicator

b. Secondary Communicator

c. Desk NCO

d. Board NCO

ANSWER: 012 (1.00)

b.

REFERENCE:

Licensed Operators/Emergency Response, 0215.000.00B-00002-02, page 30, LO 11.

[3.1/4.4*]

194001A116 ..(KA's)

QUESTION: 013 (1.00)

The oncoming NCO should complete the turnover checklist within how many hour(s) of shift turnover.

A. one-half

B. one

C. two

D. three

ANSWER: 013 (1.00)

c.

REFERENCE:

Facility Exam Bank 000762
Shift Turnover Responsibilities, SC.OP-DD.ZZ-OD20, page 16, Rev. 1.
Shift Turnover and Log Keeping, AR7000-02, Learning Objective 1.

[2.5/3.4]

194001A103 ..(KA's)

QUESTION: 014 (1.00)

Control rods are in automatic. Compare the output signal from the reactor control unit on a 2% step change in turbine load with the power at 90% with a 2% step change at 40% power.

At 90% power the signal would be:

a. the same as at 40% power.

b. larger due to the response of the Non-Linear gain unit.

c. smaller due to the response of the Non-Linear gain unit.

d. smaller due to the response of the Variable gain unit.

ANSWER: 014 (1.00)

d.

REFERENCE:

Rod Control System Description, 300S-Misc/2:25, Page 14.
Rod Control System Lesson, ROC000-01, LO 2.e.

[3.5/3.8]

001000K403 ..(KA's)

QUESTION: 015 (1.00)

Control rods are being withdrawn using the MAN position for a reactor startup with Control Bank C at 28 steps.

Select the expected position of Control Bank B.

a. 100 steps

b. 128 steps

c. 156 steps

d. 228 steps

ANSWER: 015 (1.00)

c.

REFERENCE:

Rod Control System Description, 300S-Misc/2:25, Page 25.
Rod Control System Lesson, ROC000-01, LO 2.b.

[2.6/2.8]

001010K404 ..(KA's)

QUESTION: 016 (1.00)

Which of the following instrument failures would result in a lowering in the Rod Insertion Limit?

a. The highest reading Thot fails downscale.

b. The highest reading Thot fails upscale.

c. The lowest reading Tavg output fails upscale.

d. The highest reading Tcold fails downscale.

ANSWER: 016 (1.00)

a.

REFERENCE:

Rcd Position Indication System Lesson Plan, 302/304-145.12-RPI-04, page 13, LO 1/2.4.

[4.2/4.4]

001010A103 ..(KA's)

QUESTION: 017 (1.00)

Control rods are in automatic and inserting due to an instrument failure. The Rod Bank LoLo Limit (OHA E-16) is received.

This indicates that the rods have reached the Rod Insertion Limit and:

a. the rods will stop and must be reset to allow further insertion.

b. the rods will continue to insert.

c. the rods will stop but will start rod insertion again when a lower rod insertion limit is computed.

d. the rods will insert only in manual control.

ANSWER: 017 (1.00)

b.

REFERENCE:

Rod Position Indication System Lesson Plan, 302/304-145.12-RPI-04, page 13, LO 1/2.6.

[3.6/3.6]

001000G008 ..(KA's)

QUESTION: 018 (1.00)

Following a Loss of Coolant Accident the Upper Range of Reactor Vessel Level Instrument System (RVLIS) is being used to monitor level. No RCPs are running

Select the expected response of Train A and Train B of Upper Range RVLIS indications on the RVLIS summary page if #22 RCP is started.

a. Both will indicate the same as before the pump is started.

b. Train A will indicate the same as before the pump is started but Train B will indicate downscale.

c. Train A will indicate downscale but Train B will indicate the same as before the pump is started.

d. Both will indicate downscale.

ANSWER: 018 (1.00)

d.

REFERENCE:

Reactor Vessel Level Instrumentation System Description, Volume 2, Chapter 24, page 34.
Reactor Vessel Level Instrumentation System Lesson Plan, RVLISO-06, Volume 8, Chapter 28, Learning Objective 7.a.

[3.4*/3.4*]

016000K101 ..(KA's)

QUESTION: 019 (1.00)

Which of the following changes in parameters will cause the overtemperature delta T setpoint to increase?

a. Tavg decreases.

b. Pressure decreases.

c. Delta I becomes more positive within the target band.

d. Delta I becomes more negative within the target band.

ANSWER: 019 (1.00)

a.

REFERENCE:

Reactor Coolant System Temperature Instrumentation System Description, Volume 2, Chapter 24a, page 13.
Reactor Coolant System Temperature Instrumentation Lesson Plan, RCTEMP-01, Volume 8, Chapter 24, LO 2.b.

[2.9*/3.4*]

012000A101 ..(KA's)

QUESTION: 020 (1.00)

With operation at 63% power the controlling channel, channel I, of pressurizer pressure fails downscale. With no operator action what will be the expected response?

a. A reactor scram and safety injection.

b. Pressure cycling from 2315 to 2335 as PR1 opens and closes.

c. Pressure cycling from 2315 to 2335 as PR2 opens and closes.

d. Pressure cycling from 2315 to 2335 as both PR1 and PR2 open and close.

ANSWER: 020 (1.00)

b.

REFERENCE:

Pressurizer Pressure and Level Control System Description, Volume 2, Chapter 25, page 39.
Pressurizer Pressure and Level Control System Lesson Plan, PXRP/L-00, Volume 7, Chapter 5, Learning Objective 2.h.

[3.6/3.5]

010000A302 ..(KA's)

QUESTION: 021 (1.00)

The reactor is shutdown and a cooldown is in progress with wide range RCS temperature at 300 F. The control bezel for PORV PR1 is selected to MANUAL. STOP VALVE PR6 is closed.

Select the response of PORV PR1 and STOP VALVE PR6 if the operator places the keylock switch for channel I of Pressurizer Overpressure Protection System (POPS) to ON.

a. PR1 will remain in MANUAL. POPS will not be enabled until the operator places PR1 in AUTOMATIC. STOP VALVE PR6 will remain closed.

b. PR1 will remain in MANUAL. POPS will NOT be enabled. STOP VALVE PR6 will open.

c. PR1 will transfer to automatic mode and POPS will be enabled. STOP VALVE PR6 will remain closed.

d. PR1 will transfer to automatic mode and POPS will be enabled. STOP VALVE PR6 will open.

ANSWER: 021 (1.00)

d.

REFERENCE:

Pressurizer Pressure and Level Control System Description, Volume 2, Chapter 25, page 17.
Pressurizer Pressure and Level Control System Lesson Plan, PZRP/L-00, Volume 7, Chapter 5, Learning Objective 2.f.

[4.0/3.8]

010000A403 ..(KA's)

QUESTION: 022 (1.00)

The following Plant conditions exist:

MODE 1 with reactor power at 19%.
Tave at 555 degrees F.
Turbine Power is 16%
Steam Dumps in AUTOMATIC in MS PRESSURE CONTROL mode.

Which ONE of the following statements describes the plant response if PT-507, Steam Header Pressure, fails HIGH?

a. Steam dumps remain closed because the arming signal is absent since the difference between reactor and turbine power is 3%.

b. Steam dumps open with arming signal present since the steam header pressure is greater than the controller setpoint.

c. Steam dumps open with arming signal present since Tave-Tref has greater than a 2 degree difference.

d. Steam dumps remain closed because the arming signal is absent since Tave is less than the LOW Tave setpoint.

ANSWER: 022 (1.00)

b.

REFERENCE:

Steam Dump TSD System Description, Volume 2, Chapter 26, page 13.
Steam Dump Lesson Plan, STDUMP-03, Volume 7, Chapter 6, Learning Objective 2.f.

[3.1/3.2]

041020A102 ..(KA's)

QUESTION: 023 (1.00)

Which ONE of the following describes operation of the Source Range (SR) Nuclear Instrumentation during startup?

a. The "BLOCK SOURCE RANGE" pushbuttons are used to turn off the SR high voltage when 1/2 Intermediate Range channels are greater than 1x10E-10 amps.

b. The "BLOCK SOURCE RANGE" pushbuttons are used to turn off the SR high voltage when 2/2 IR channels are between 1x10E-11 and 1x10E-10 amps.

c. The SR high voltage is automatically blocked when 1/2 IR channels exceeds P-10.

d. The SR high voltage is automatically blocked when 2/2 IR channels exceed P-6.

ANSWER: 023 (1.00)

a.

REFERENCE:

Excore Nuclear Instrumentation System Description, Volume 2, Chapter 20, page 20.
Excore Nuclear Instrumentation Lesson Plan, EXCNIS-07, Volume 7, Chapter 01, Learning Objective 2.f.

[3.1/3.3]

015000K401 ..(KA's)

QUESTION: 024 (1.00)

A power range channel fails high with rod control in manual. Which of the following actions will occur?

a. PR Positive rate reactor trip.

b. PR High Flux - High level reactor trip.

c. Overpower rod stop.

d. Over Power delta T turbine runback.

ANSWER: 024 (1.00)

c.

REFERENCE:

Excore Nuclear Instrumentation System Description, Volume 2, Chapter 20, page 27.
Excore Nuclear Instrumentation Lesson Plan, EXCNIS-07, Volume 7, Chapter 01, Learning Objective 3.1.

[3.7/3.9]

015000K402 ..(KA's)

QUESTION: 025 (1.00)

During Mode 5 operations, the operators receive the following alarms/indications:

- Audible count rate increase
- High Flux at Shutdown alarm
- Source Range indicators increasing

Select the ONE response below which describes the proper actions to be taken for the given conditions.

a. Immediately open the trip breakers and initiate boration

b. Make a PA announcement to all personnel to evacuate containment, calculate SDM.

c. Place both Primary Water pumps to off and verify all dilution paths are isolated.

d. Verify both Primary Water pumps are off, close CV179, and rapid borate.

ANSWER: 025 (1.00)

b.

REFERENCE:

S2.OP-AR.ZZ-0005
TSD Chapter 20.
Facility Exam Bank 000056

[3.8/3.8]

015000G001 ..(KA's)

QUESTION: 026 (1.00)

With one bypass breaker closed, racking the second bypass breaker in will result in:

a. opening all trip and bypass breakers due to a General Warning Reactor Trip.

b. opening only the closed bypass breaker.

c. opening the trip breaker in parallel with the closed bypass breaker only.

d. a General Warning Alarm in the control room but not open any trip or bypass breakers.

ANSWER: 026 (1.00)

a.

REFERENCE:

Reactor Protection System Description, Volume 3, Chapter 28, page 17.
Reactor Protection System Lesson Plan, RXPROT-08, Volume *, Chapter 2, Learning Objective 3.

[3.3/3.6]

012000K604 ..(KA's)

QUESTION: 027 (1.00)

Which of the following events can the operator use the MANUAL RESET pushbuttons to restore feedwater capability to the steam generator?

a. A high steam line flow causes a low Tavg and an SI.

b. A steam dump fails open on a turbine trip from 65% power causing a low Tavg and reactor trip.

c. The operator overfeeds a single steam generator to the High-High Level.

d. A spray valve fails open causing RCS pressure to decrease to 1725 psig.

ANSWER: 027 (1.00)

b.

REFERENCE:

Condensate and Feedwater System Description, Volume 3, Chapter 34, page 20.
Condensate and Feedwater System Lesson Plan, CN/BFO-05, Volume 3, Chapter 18, Learning Objective 2.f.

[3.1/3.3]

059000A411 ..(KA's)

QUESTION: 028 (1.00)

A spurious Phase "A" signal has occurred.

Identify the effect this will have on RCP operation.

a. The RCP seal leakoff will be isolated. The seals will be directed to the PRT via a relief valve.

b. The CCW to the RCP thermal barrier will be lost so seal injection must be maintained.

c. The CCW to the RCP motor will be lost so the RCP's will have to be tripped.

d. The RCP seal injection is isolated, so CCW flow to the thermal barrier must be verified.

ANSWER: 028 (1.00)

a.

REFERENCE:

Reactor Coolant Pump System Description, Volume 1, Chapter 4, page 16.
Reactor Coolant Pump Lesson Plan, RCPUMP-04, Volume 1, Chapter 6, Learning Objective 15.

[3.3/3.6]

003000K103 ..(KA's)

QUESTION: 029 (1.00)

The reactor is shutdown with all reactor coolant pumps stopped. A steam bubble exists in the pressurizer. It is desired to start a RCP.

Which RCP would be most effective in providing pressure control?

a. 21

b. 22

c. 23

d. 24

ANSWER: 029 (1.00)

c.

REFERENCE:

Reactor Coolant Pump System Description, Volume 1, Chapter 4, page 18
Reactor Coolant Pump Lesson Plan, RCPUMP-04, Volume 1, Chapter 6, Learning Objective 13.

[3.6/3.7]

010000K103 ..(KA's)

QUESTION: 030 (1.00)

Leakage into #13 steam generator is determined to be 0.5 gpm. No leakage is detectable into the other steam generators. Other leakage which cannot be identified is determined to be 0.6 gpm. Leakage from known sources other than steam generator leakage is determined to be 4.0 gpm.

With these conditions in existence, Technical Specification leakage limits:

a. are not exceeded.

b. are exceeded due to the total leakage into the steam generator and unidentified leakage exceeding 1 gpm.

c. are exceed due to steam generator leakage exceeding limits for pressure boundary leakage.

d. are exceeded due to excessive leakage into one steam generator.

ANSWER: 030 (1.00)

d.

REFERENCE:

Salem Unit #1 Technical Specifications, Section 3.4.6.2.
Steam Generator Lesson Plan, STMGEN-02, Volume 1, Chapter 7, Learning Objective 13.

[3.6/4.1]

002000G005 ..(KA's)

QUESTION: 031 (1.00)

The reactor is operating at 100% power steady state. A 75 gpm letdown orifice is in service. Pressurizer level is constant.

The expected charging header flowrate at the discharge of the charging pumps would be:

a. 63 gpm.

b. 75 gpm.

c. 87 gpm.

d. 95 gpm.

ANSWER: 031 (1.00)

c.

REFERENCE:

Chemical and Volume Control System Description, Volume 1, Chapter 6, Figure CV-2.
Chemical and Volume Control System Lesson Plan, CVCS00-03, Volume 1, Chapter 8, Learning Objective 2.

[3.0/3.0]

004020A108 ..(KA's)

QUESTION: 032 (1.00)

The Unit 2 CVCS Makeup Mode Selector switch is selected to AUTOMATIC.

Identify the response if VCT level decreases to the setpoint for automatic makeup.

a. Boron flow rate is set at a specific rate by the operator and Primary water flow rate is set at 70 gpm. Flow is via CV-181 to the VCT.

b. Boron flow rate is set at a 40 gpm and the primary water flow rate is set at a specific rate by the operator. Flow is via CV-181 to the VCT.

c. Boron flow rate is set at a specific rate by the operator and Primary water flow rate is set at 70 gpm. Flow is via CV-185 to the Charging Pumps suction.

d. Boron flow rate is set at a 40 gpm and the primary water flow rate is set at a specific rate by the operator. Flow is via CV-185 to the Charging Pumps suction.

ANSWER: 032 (1.00)

c.

REFERENCE:

Chemical and Volume Control System Description, Volume 1, Chapter 6, page 36.
Chemical and Volume Control System Lesson Plan, CVCS00-03, Volume 1, Chapter 8, Learning Objective 2.e.

[3.8/3.3]

004020A401 ..(KA's)

QUESTION: 033 (1.00)

Select the action that prevents discharging water from the containment sump to the RWST via the SI pump mini-flow lines on Unit 1.

a. Mini-flow valves SJ67 and SJ68 close when SI pumps suction SJ45 is opened.

b. The power lockout switch must be placed to VALVE OPERABLE position which causes the mini-flow valves to close.

c. SJ67 and SJ68 valve control must be placed to RECIRC OVERRIDE CLOSE which causes the mini-flow valves to be overridden closed.

d. SI pump suction SJ45 is interlocked closed until either mini-flow valves is closed.

ANSWER: 033 (1.00)

d.

REFERENCE:

Emergency Core Cooling System Description, Volume 1, Chapter 10, page 16.
Emergency Core Cooling Lesson Plan, ECCS00-06, Volume 1, Chapter 10, Learning Objective 7.
NOTE: Facility needs to verify correct valve terminology for unit 1 valves.
[3.6*/3.7]

006000K410 ..(KA's)

QUESTION: 034 (1.00)

Select the expected position of the steam generator inlet control valves (AF-21s) prior to startup of a motor driven auxiliary feedwater pump and the reason for the valves to be in that position.

a. Open because the demand signal is set for 95% by the operator.

b. Closed because the PRESSURE OVERRIDE function is in effect.

c. Open because the valve is interlocked open when the motor breaker is open.

d. Closed because the valve is interlocked closed when the motor breaker is closed.

ANSWER: 034 (1.00)

b.

REFERENCE:

Auxiliary Feedwater System Description, Volume 1, Chapter 11, page 20.
Auxiliary Feedwater System Lesson Plan, AFEEDW-10, Volume 1, Chapter 14, Learning Objective 2.e

[3.1/3.4.]

061000K404 ..(KA's)

QUESTION: 035 (1.00)

The AFW pump low suction pressure trips are made operable during which of the following conditions.

a. Severe weather.

b. Local Operation of the AFW pumps.

c. Alignment of AFW to an alternate suction source.

d. Potential exists for freezing of the AFWST.

ANSWER: 035 (1.00)

a.

REFERENCE:

Auxiliary Feedwater System Description, Volume 1, Chapter 11, table 2.
Auxiliary Feedwater System Lesson Plan, AFEEDW-10, Volume 1, Chapter 14, Learning Objective 4.a.

[3.1*/3.3*]

061000K407 ..(KA's)

QUESTION: 036 (1.00)

During a loss of coolant accident the containment spray system initiates but fails to inject NaOH.

Select the expected condition resulting from the failure to inject NaOH.

a. Containment sump will be acidic and increased potential for higher dose during a release from iodine.

b. Containment sump will be acidic and increased potential for hydrogen explosion.

c. Containment sump will be basic and increased potential for higher dose during a release from iodine.

d. Containment sump will be basic and increased potential for hydrogen explosion.

ANSWER: 036 (1.00)

a.

REFERENCE:

Containment and Support Systems System Description, Volume 1, Chapter 12, page 17.
Containment Spray Lesson Plan, 300S-000.0S-CSPRAY-06, Volume 1, Chapter 11, Learning Objective 1.

[3.6/3.9]

026000G004 ..(KA's)

QUESTION: 037 (1.00)

A steam break occurred inside containment concurrent with an undervoltage condition on the 4KV vital busses. Prior to the containment pressure reaching the high high containment pressure setpoint, SEC is reset but SI is NOT reset.

Based on the above conditions, select the response of the containment spray system when containment pressure reaches the high high containment pressure setpoint.

a. No action will occur.

b. The containment spray valves will reposition but the pumps will NOT start.

c. The containment spray valves will NOT reposition but the pumps will start.

d. The containment spray valves will reposition and the pumps will start.

ANSWER: 037 (1.00)

b.

REFERENCE:

Containment Spray System Description, Volume 1, Chapter 12A, page 8.
Containment Spray System Lesson Plan, Volume 1, Chapter 11, Learning Objective 2.f.

[4.3/4.5]

026000A301 ..(KA's)

QUESTION: 038 (1.00)

A Safety Injection has occurred. During bus loading an undervoltage condition occurs simultaneously on the "2A" and "2B" vital busses.

Select the mode response of SEC for all vital busses.

a. Busses "2A" and "2B" will go through Mode III sequencing; Bus "2C" will go through Mode I sequencing.

b. Busses "2A" and "2B" will go through Mode II sequencing; Bus "2C" will go through Mode I sequencing.

c. All busses will go through a Mode III sequencing.

d. All busses will go through a Mode II sequencing.

ANSWER: 038 (1.00)

c.

REFERENCE:

Safeguards Equipment Control System Description, Volume 1, Chapter 13, page 5.
Safeguards Equipment Control Lesson Plan, SEC000-01, Volume 6, Chapter 19, Learning Objective 2. Page 32.

[3.2/3.8]

013000K411 ..(KA's)

QUESTION: 039 (1.00)

Following a Mode II operation of Safeguards Equipment Control (SEC), when will the 230 V Control Centers lockouts automatically reset?

a. 5 minutes.

b. 10 minutes.

c. 20 minutes.

d. 30 minutes.

ANSWER: 039 (1.00)

c.

REFERENCE:

Safeguards Equipment Control Lesson Plan, SEC000-01, Volume 6, Chapter 19, page 15, Learning Objective 2.f.

[3.6*/3.7*]

064000A307 ..(KA's)

QUESTION: 040 (1.00)

The plant is operating at 65% power with No. 11 and No. 12 Component Cooling Water(CCW) pumps selected to MANUAL and START. No. 13 CCW is selected to AUTO. The operator observes that No. 13 CCW pump "Start" pushbutton backlight is flashing. The "Stop" pushbutton backlight is not flashing.

Select the condition that will cause this to occur?

a. Pump control has been transferred to the Hot Shutdown Panel.

b. The pump has auto started on low system pressure.

c. Control of the pump has been transferred to manual by the SEC.

d. The breaker failed to close when told to by the SEC.

ANSWER: 040 (1.00)

b.

REFERENCE:

Component Cooling Water System Description, Volume 1, Chapter 14, page 8.
Component Cooling Water Lesson Plan, CCW000-02, Volume 1, Chapter 12, Learning Objective 2.d.

[3.3/3.1]

008000A401 ..(KA's)

QUESTION: 041 (1.00)

A diesel generator is operating and carrying its Vital Bus due to SEC operation.

Which of the following conditions will trip the diesel?

a. Overspeed from the electrical overspeed circuit.

b. High Lube Oil Temperature.

c. High Jacket Water Temperature.

d. Loss of Generator Excitation.

ANSWER: 041 (1.00)

a.

REFERENCE:

Diesel Generator and Support Systems System Description, Volume 4, Chapter 41, page 48.
Diesel Generator and Support Systems Lesson Plan, DIESEL-05, Volume 5, Chapter 09, Learning Objective 5.

[3.9/4.2]

064000K402 ..(KA's)

QUESTION: 042 (1.00)

Which of the following Control Area Ventilation operational condition lineups occurs automatically?

a. Fire Inside Control Area.

b. Fire Outside Relay Area.

c. Accident-Inside Air

d. Accident-Mixed Air

ANSWER: 042 (1.00)

c.

REFERENCE:

Control Area Ventilation System Description, Volume 4, Chapter 48, Table II.
Control Area Ventilation Lesson Plan, CAVENT-02, Volume 2, Chapter 11, Learning Objective 7.

[2.8/3.1]

013000K113 ..(KA's)

QUESTION: 043 (1.00)

Which one of the following conditions will cause the automatic start of both diesel drive fire pumps?

a. Fire main pressure decreases to 80 psig.

b. Loss of normal AC power.

c. The jockey pump trips.

d. Fire main pressure decreasing at a rate faster than 10 psi/sec.

ANSWER: 043 (1.00)

b.

REFERENCE:

Fire Protection System Description, Volume 4, Chapter 50, page 30.
Fire Protection Lesson Plan, FIRPRO-03, Volume 4, Chapter 30, Learning Objective 2.e.

[3.0/3.3]

086000A202 ..(KA's)

QUESTION: 044 (1.00)

Select the statement that describes the initiation logic for AMSAC. Assume power is at 85%.

a. 1/1 low low level on 2/4 steam generators for 25 seconds.

b. 2/3 low low level on 2/4 steam generators for 25 seconds.

c. 1/1 low low level on 3/4 steam generators for 25 seconds.

d. 2/3 low low level on 3/4 steam generators for 25 seconds.

ANSWER: 044 (1.00)

c.

REFERENCE:

AMSAC System Description, Volume 4 (cont), Chapter 53, page 15.
ATSW Mitigation System Actuation Circuitry (AMSAC) Lesson Plan, AMSAC0-00, Volume 8, Chapter 9, Learning Objective 5.

[4.5/4.6]

061000K402 ..(KA's)

QUESTION: 045 (1.00)

Three containment fan coil units are operating in FAST prior to an SI signal. All containment fain coil units are operable.

Select the expected TOTAL service water flow to containment fan coil units prior to and after the SI signal is received.

a. Before - 7650; After - 3750

b. Before - 2250; After - 12,750.

c. Before - 2250; After - 7650

d. Before - 12,750; After - 3750

ANSWER: 045 (1.00)

b.

REFERENCE:

Service Water System Description, Volume 2, Chapter 15, page 37.
Service Water Lesson Plan, SWATER-08, Volume 1, Chapter 13, Learning Objective 21

[3.7/3.7]

076000A302 ..(KA's)

QUESTION: 046 (1.00)

With the plant at full power, a reactor trip signal is generated through the Solid State Protection System (SSPS) due to a Instrumentation Technician error. On the trip, the Train A Reactor Trip Breaker failed to open. Assuming no operator action is taken, select the consequence of this breaker failure.

a. No adverse consequences to the steam dump system. A single failure of an arming signal input has no affect.

b. RCS temperature will be driven to 547F via the Plant Trip Controller.

c. RCS temperature will be driven to 552F via the Load Rejection Controller instead of the Plant Trip Controller.

d. Steam dumps will not arm and RCS temperature will be maintained via the MS10's and SG safeties.

ANSWER: 046 (1.00)

b.

REFERENCE:

Facility Question Bank 000582.
Steam Dump System Description, Volume 2, Chapter 26, page 20.
Steam Dump Lesson Plan, STDUMP-03, Volume 7, Chapter 6, Learning Objective 2.

[3.8/3.9]

041020K302 ..(KA's)

QUESTION: 047 (1.00)

Given the following:

- Reactor power is at 25%.
- All plant controls are in automatic.
- Loop 21 MSIV inadvertently shuts.

WHICH ONE (1) of the following parameters would show an INITIAL INCREASE following this event? (Assume NO operator actions are taken).

a. Steam generator #21 level.

b. Steam Generator #22 pressure.

c. Loop #24 Cold leg temperature.

d. Steam Generator #23 level.

ANSWER: 047 (1.00)

d.

REFERENCE:

Facility Question 001346.
Steam Generator Water Level Control System Description, Volume 2, Chapter 27, page 7.
Steam Generator Water Level Control System Lesson Plan, Volume 7, Chapter 08, Learning Objective 8.

[2.8/3.1]

035010K503 ..(KA's)

QUESTION: 048 (1.00)

The group demand counters for Control Bank D are at 100 steps.

In order to satisfy Technical Specification limits, individual rod positions for rods in Control Bank D must indicate from:

a. 88 to 100 steps.

b. 100 to 112 steps.

c. 88 to 112 steps.

d. 96 to 104 steps.

ANSWER: 048 (1.00)

c.

REFERENCE:

Rod Position Indication System Lesson Plan, 302S-RPI-04, Volume 7, Chapter 4, Learning Objective 3.0.
Technical Specifications 3.1.3.1.

[3.1/3.7]

014000G005 ..(KA's)

QUESTION: 049 (1.00)

The Unit is at full power when the 115 VAC infeed breaker to the Output Bay of SSPS train 'A' inadvertently opens.

This event will:

a. cause a reactor trip.

b. cause an SI actuation.

c. have no effect since this power supply is auctioneered with 115 VAC from the Logic Bay.

d. prevent an SI actuation from Train 'A'.

ANSWER: 049 (1.00)

d.

REFERENCE:

Facility Question Bank 000772
Reactor Protection System Description, Volume 3, Chapter 28, page 21.
Reactor Protection System Lesson Plan, RXPROT-08, Volume 8, Chapter 2, Learning Objective 2.g.

[3.3/3.7]

012000K201 ..(KA's)

QUESTION: 050 (1.00)

Select the mechanism used by the 115 V Vital Power to transfer power to the 125 VDC power supply.

a. A power auctioneering circuit transfers power from the Normal AC Input to the 125 V DC supply.

b. A static switch transfers power from 230 V AC emergency power to 125 V DC supply.

c. When low voltage is sensed from the 230 V AC Emergency Power the 125 V DC input breaker is automatically shut.

d. A power auctioneering circuit transfers power from the 230 V DC Emergency Power to 125 V DC.

ANSWER: 050 (1.00)

a.

REFERENCE:

Electrical Distribution System Description, Volume 4, Chapter 42D, page 2.
115 VAC Control Power System Lesson Plan, Volume 5, Chapter 08, Learning Objective 2.e.

[3.1/3.5]

062000K410 ..(KA's)

QUESTION: 051 (1.00)

A loss of off-site power has occurred. The operator observes that 2A Diesel is operating at 2950 KW.

Select the MINIMUM required action for continued operation of 2A diesel generator.

a. Reduce load to less than 2860 immediately.

b. Reduce load to less than 2860 within 30 minutes.

c. Reduce load to less than 2860 within 2 hours.

d. Reduce load to less than 2750 within 30 minutes.

ANSWER: 051 (1.00)

b.

REFERENCE:

EDG & Support Systems Lesson Plan, DIESEL-05, Volume 5, Chapter 09, Page 40, Learning Objective 12.a

[3.1/3.2]

064000G010 ..(KA's)

QUESTION: 052 (1.00)

Which of the following describes the purpose of the FIRE EMERGENCY BYPASS switches located on the local control panel?

a. Bypass control room controls.

b. Bypass the SEC load sequencing signals.

c. Bypass control room and cable room wiring to allow local starting of the diesel during a shutdown outside the control room.

d. Bypass Carbon Dioxide fire protection for the diesel generator in order to test fire detectors.

ANSWER: 052 (1.00)

b.

REFERENCE:

Diesel Generator and Support Systems System Description, Volume 4, Chapter 41, page 38.
Diesel Generator and Support Systems Lesson Plan, DIESEL-05, Volume 5, Chapter 09, Learning Objective 14.

[3.4/3.5]

064000G007 ..(KA's)

QUESTION: 053 (1.00)

A RCP has just been started with RCS pressure at 325 psig. Select the reason that a minimum of 200 psid on the RCP seals is required for RCP operation?

a. Ensures that adequate seal cooling flow from the RCS is available.

b. Prevents the #1 RCP seal from swapping from a face rubbing to a film riding seal.

c. Prevents the weight of the seal ring from limiting cooling flow through the seal gap.

d. To avoid having to open the bypass valve around #1 seal.

ANSWER: 053 (1.00)

c.

REFERENCE:

Facility Exam Bank 001334
Reactor Coolant Pump System Description, Volume 1, Chapter 4, page 15.
Reactor Coolant Pump Lesson Plan, RCPUMP-04, Volume 1, Chapter 6, Learning Objective 13.

[2.8/3.1]

003000K404 ..(KA's)

QUESTION: 054 (1.00)

A reactor cooldown is in progress and the Low Steam Line Pressure SI signal has been manually blocked.

Identify the plant response in the event of a main steam line rupture at the mixing bottle.

a. Both an Safety Injection and Main Steam Line Isolation will occur.

b. A Main Steam Line Isolation will occur.

c. A Main Steam Line Isolation will occur only if pressure decreases to below 600 psig.

d. An Safety Injection will occur but a Main Steam Line Isolation will NOT occur.

ANSWER: 054 (1.00)

b.

REFERENCE:

Reactor Protection System Description, Volume 3, Chapter 28, page 40.
Reactor Protection System Lesson Plan, RXPROT-08, Volume 8, Chapter 2, Learning Objective 2.

[3.9/4.4]

013000K403 ..(KA's)

QUESTION: 055 (1.00)

Which of the following mode changes for the Subcooling Margin Monitor (SMM) will ONLY occur manually?

a. Shifting from "Normal" to "Adverse" mode on increasing containment pressure.

b. Shifting from "Adverse" to "Normal" mode on decreasing containment pressure.

c. Shifting from "Normal" to "Adverse" mode on increasing containment radiation.

d. Shifting from "Adverse" to "Normal" mode on decreasing containment radiation.

ANSWER: 055 (1.00)

d.

REFERENCE:

Facility Exam Bank 000760
Incore Nuclear Instrumentation System Description, Volume 2, Chapter 21, page 26.
Incore Nuclear Instrumentation System Lesson Plan, INCORE-04, Volume 7, Chapter 02, Learning Objective 5.

[2.9/2.9]

017000G007 ..(KA's)

QUESTION: 056 (1.00)

Operating procedure S2.OP-SO.RM-0001, Radiation Monitoring System - Normal Operation contains a precaution which specifies that use of the installed check source is not to exceed 30 seconds.

Select the basis for this precaution.

a. Avoid "masking" of the current radiation levels.

b. Avoid actuation of the amber "FAILURE" light, terminating interlock functions.

c. Prevent overheating of the check source actuation solenoid.

d. Avoid detector damage due to detector burnout.

ANSWER: 056 (1.00)

c.

REFERENCE:

Facility Exam Bank 000999
Radiation Monitoring System, S2.OP-SO.RM-00001, Step 3.2.
Radiation Monitoring System Lesson Plan, RMS000-00, Volume 8, Chapter 12, Learning Objective 12.

[2.8/3.0]

072000G010 ..(KA's)

QUESTION: 057 (1.00)

Which of the following would be indication that VCT level detector LT-114 has failed low?

a. VCT Hi-Lo Level alarm is annunciated.

b. Following a down power transient, CV-35 trips open instead of modulating open.

c. Auto Makeup operates continuously.

d. VCT Level indication on the control console indicates downscale.

ANSWER: 057 (1.00)

b.

REFERENCE:

Chemical and Volume Control System Description, Volume 1, Chapter 6, page 22-23.
Chemical and Volume Control System Lesson Plan, CVCS00-03, Volume 1, Chapter 8, Learning Objective 2.e.

[3.2/3.0]

004020A305 ..(KA's)

QUESTION: 058 (1.00)

The Gamma Metrics are provided for post accident indication in:

a. only the source range.

b. only the source and intermediate range.

c. only the power range.

d. all ranges of NIs.

ANSWER: 058 (1.00)

d.

REFERENCE:

Excore Nuclear Instrumentation System Description, Volume 2, Chapter 20, page 35.
Excore Nuclear Instrumentation Lesson Plan, EXCNIS-07, Volume 7, Chapter 01, Learning Objective 2.

[3.9/3.9]

015000A402 ..(KA's)

QUESTION: 059 (1.00)

WHICH ONE (1) of the following will cause an AUTOMATIC trip of a Containment Iodine Removal System Fan?

a. Phase A

b. Safety Injection signal

c. Roughing Filter HIGH differential pressure

d. Charcoal Filter HIGH temperature

ANSWER: 059 (1.00)

b.

REFERENCE:

Facility Exam Bank 001342
Containment and Support Systems Lesson Plan, CONTMT-02, Volume 2, Chapter 01, Page 59, Learning Objective 2.f.

[3.3*/3.2*]

027000A403 ..(KA's)

QUESTION: 060 (1.00)

Following a loss of all off-site power, EDG "2C" fails to start.

Which of the following lists BOTH the control air headers that will NOT have a source of air?

a. 2A and 1A

b. 2A and 2B

c. 1A and 1B

d. 1B and 2B

ANSWER: 060 (1.00)

a.

REFERENCE:

Station and Control Air System Description, Volume 3, Chapter 39, page 13.
Control Air Lesson Plan, 300S-CONAIR-07, Volume 2, Chapter 03, Learning Objective 15.

[3.0/3.4]

078000K303 ..(KA's)

QUESTION: 061 (1.00)

Temperature of the air mixture in the containment hydrogen recombiners is controlled by:

a. restricting the inlet air flow to the recombiner.

b. controlling the air flow at the discharge of the recombiner which is used to preheat the inlet flow.

c. cycling the electric heaters on and off to maintain temperature.

d. adjusting the power to the electric heaters.

ANSWER: 061 (1.00)

d.

REFERENCE:

Containment and Support Systems Lesson Plan, CONTMT-02, Volume 2, Chapter 01, page 111, Learning Objective 2.b

[2.6/3.1]

028000K601 ..(KA's)

QUESTION: 062 (1.00)

A small break loss of coolant accident has occurred inside containment. Select the effect that an increase in containment temperature will have on pressurizer level.

Reference leg density will decrease causing:

a. decreased differential pressure and an increase in level indication.

b. increased differential pressure and an increase in level indication.

c. decreased differential pressure and a decrease in level indication.

d. increased differential pressure and a decrease in level indication.

ANSWER: 062 (1.00)

a.

REFERENCE:

Pressurizer Pressure and Level Control, PZRP/L-00, Volume 7, Chapter 05, page 36, Learning Objective 2.

[3.5/3.6]

011000A101 ..(KA's)

QUESTION: 063 (1.00)

A residual heat removal pump is being used for shutdown cooling. The pump is at operating temperature.

If the pump trips what is the restriction on restarting the pump.

The pump can be started:

a. only after investigation using a no plan work order.

b. four times an hour.

c. Once; the pump must remain at a standstill for 20 minutes before another start is attempted.

d. Twice; following the second attempt the pump must remain at a standstill for 45 minutes before another start is attempted.

ANSWER: 063 (1.00)

a.

REFERENCE:

Circuit Breaker Reclosure Policy Following a Trip, Operation Directive OD-06.
Large Motor Starting Criteria, SC.OP-DD.ZZ-OD03, rev. 0, page 1.

[3.3/3.5]

005000G010 ..(KA's)

QUESTION: 064 (1.00)

During a startup, the High Neutron Flux Reactor Trip and the Control Rod Stop functions on the Intermediate Range Nuclear Instruments are blocked by depressing a pair of pushbuttons on the control board.

How does this affect operation of the channel?

a. The input to the Level trip and Rod Stop bistables are shunted to ground.

b. The output from the Level Trip and Rod Stop bistables is blocked.

c. A bypass circuit around the Level Trip and Rod Stop bistables is shunted closed.

d. The input from the detector to the log current amplifier is blocked.

ANSWER: 064 (1.00)

b.

REFERENCE:

Excore Nuclear Instrumentation Lesson Plan, EXCNIS-07, Volume 7, Chapter 01, Learning Objective 2.f., page 44.

[3.9/4.2]

015000K406 ..(KA's)

QUESTION: 065 (1.00)

From 50% power, a control bank rod drops without causing a reactor trip. The procedure for recovering the rod requires the dropped rod's group demand step counter indication be recorded prior to starting the recovery of the rod.

Which one of the following describes why this reading is recorded?

a. To verify the operability of the dropped rod after recovery.

b. To track that the rod insertion limits are not been violated during recovery.

c. To insure the bank overlap unit is reset to its proper value after recovery.

d. To insure the operator positions the dropped rod to the correct height after recovery.

ANSWER: 065 (1.00)

d.

REFERENCE:

Salem, S2.OP-AB.ROD-0002 Rev.02
Salem LP ABROD02-01, EO 3; III.C.1, page 5.
Facility Bank Mod (000289).
KA [3.0/3.5]

000003K309 ..(KA's)

QUESTION: 066 (1.00)

Given the following for Unit 2:

The reactor has tripped.
3 control rods have failed to insert on the trip.

Which one of the following is the MINIMUM volume that must be added to the RCS from the RWST as a result of the stuck rods?

a. 1960 gallons

b. 5600 gallons

c. 11200 gallons

d. 16800 gallons

ANSWER: 066 (1.00)

d.

REFERENCE:

3 rods X 80 min/rod X 70 gpm = 16,800 gal.
S2.EOP-TRIP-2, Rev. 3, sheet 1.
Salem LP TRIP02-03, EO 2.a; II.G, page 7.
Facility Bank Mod (001359).
KA [3.5/4.4]

000005A203 ..(KA's)

QUESTION: 067 (1.00)

When raising reactor power, a Control Bank C (CBC) Group 1 rod stops moving at 218 steps and the URGENT FAILURE alarm actuates.

What problem is generated if the operator continues to withdraw rods using individual bank positions (CBC & CPD) on the Bank Selector Switch?

a. The position difference between the affected rod(s) and the remainder of rods in the bank will exceed allowed Technical Specification limits.

b. The RIL Monitor will calculate a higher setpoint for the RIL alarm.

c. A loss of bank overlap in the control groups will occur.

d. The ROD BOTTOM/ROD DROP alarm will NOT clear.

ANSWER: 067 (1.00)

c.

REFERENCE:

S2.OP-AB.ROD-0001(Q) Rev. 2, Technical Bases Document, 2.4 2nd para.
Step 3.5, page 4.
Salem LP ABROD1-01, EO 3; C.1, page 6.
KA [3.6/3.4]

000005A101 ..(KA's)

QUESTION: 068 (1.00)

Given the following for Unit 2:

A small break LOCA has occurred.
The reactor is tripped with safety injection activated.
All ECCS equipment is operating as expected.
All preceding emergency procedure action steps have been performed and currently actions are being directed in 2-EOP-LOCA-2, "POST LOCA COOLDOWN AND DEPRESSURIZATION".

Which one of the following describes actions prescribed in 2-EOP-LOCA-2 which would prevent an Inadequate Core Cooling Condition?

a. Stabilize SG pressure and level, increase ECCS injection and RCS pressure to increase heat removal through the break.

b. Increase heat removal via the SGs to increase cooldown and depressurization of the RCS allowing increased ECCS flow.

c. Reduce ECCS flow and lower RCS pressure to increase heat removal through the break while maintaining a constant SG pressure.

d. Maintain pressurizer level greater than 20% and RCS pressure greater than 1000 psig to ensure subcooling can be maintained while steaming the SGs.

ANSWER: 068 (1.00)

b.

REFERENCE:

Salem LP LOCA02-02, EO 2.b; II.E & G, pages 7-8.
Facility Bank Mod (001381).
KA [4.5/4.7]

000011A210 ..(KA's)

QUESTION: 069 (1.00)

Which one of the following describes why the seal leakoff on an RCP is isolated when a No. 1 Seal Failure occurs?

a. Precludes overheating the CCW in the Thermal Barrier HX for the other RCPs.

b. Converts the No. 2 seal to a film-riding seal for the affected RCP.

c. Reduces flashing in the common seal leakoff return line.

d. Prevents overpressurizing the VCT.

ANSWER: 069 (1.00)

b.

REFERENCE:

S2.OP-AB.RCP-0001(Q) Rev. 2; 2.4, page 4.
Salem LP ABRCP1-01, EO 3 & 4A; C.1.c, page 7.
KA [3.1/3.2]

000015G007 ..(KA's)

QUESTION: 070 (1.00)

The reactor is at 100% power. All charging flow is lost and the crew is unable to restore a charging pump. During the attempts to restore charging, an operator inadvertently misunderstands an order and closes valve CC131. The valve fails to reopen and the SRO enters AB.CC-0001 "COMPONENT COOLING WATER ABNORMALITY".

Which of the following statements describes the general actions required by this procedure?

a. If RCP motor and pump bearing temperatures exceed 175 degrees F, reduce power to less than 36% and stop all RCPs.

b. If seal flow cannot be restored within 5 minutes, trip the reactor and stop all RCPs.

c. Immediately trip the reactor and stop all RCPs.

d. Reduce power to less than 36% and stop all RCPs.

ANSWER: 070 (1.00)

c.

REFERENCE:

S2.OP-AB.CC-0001 Rev.01, Step 3.1
Salem LP ABCC01-01, EO 2; III.C.1, page 5.
Facility Bank Mod(001474).
KA [3.3/3.3]

000026G012 ..(KA's)

QUESTION: 071 (1.00)

Which of the following describes the MINIMUM actions that are taken if a reactor trip is NOT confirmed at Step 5 of 2-EOP-TRIP-1 BEFORE transitioning to 2-EOP-FRSM-1 "RESPONSE TO NUCLEAR POWER GENERATION"?

1. Open Rapid Borate Stop Valve 2CV175.
2. Verify Dilution Paths are Isolated.
3. Open BOTH Reactor Trip Breakers.
4. Verify Automatic Rod Insertion.
5. Verify Feedwater Isolation.
6. Start At Least ONE Boric Acid Pump in "MANUAL-FAST".

a. 1, 3, and 6

b. 1, 3, 4 and 6

c. 1, 2, 3, 4, and 5

d. 1, 2, 3, 4, 5, and 6

ANSWER: 071 (1.00)

a.

REFERENCE:

S2.EOP-TRIP-1 Rev.03
Salem LP TRIP01-05, EO 2; V.A.4, page 8.
Facility Bank Mod (001195).
KA [4.5/4.5]

000029G010 ..(KA's)

QUESTION: 072 (1.00)

2-EOP-FRSM-1 "RESPONSE TO NUCLEAR POWER GENERATION" directs the operator to reduce RCS pressure using pressurizer PORV's if pressure is above 2335 psig. Which one of the following describes the purpose for reducing RCS pressure.

a. To allow increased boration rate from the charging system.

b. To prevent a high pressure inadequate core cooling situation.

c. To prevent a small break LOCA situation from developing from a failed open pressurizer code safety valve.

d. To allow maximum negative reactivity addition from voiding.

ANSWER: 072 (1.00)

a.

REFERENCE:

S2.EOP-FRSM-1 Rev.03
Salem LP FRSM01-01, EO 2; II.D.1, page 9.
Facility Bank Mod(000896).
KA [4.4/4.7]

000029K312 ..(KA's)

QUESTION: 073 (1.00)

In the EOP network, if RCS pressure is less than 1355 psig AND either a CCP or SI pump is running and providing at least 100 GPM, then all RCPs are stopped.

Which one of the following describes the BASIS for this step?

a. To protect the RCP seals.

b. To minimize mass loss from a small break LOCA.

c. To assist establishment of natural circulation.

d. To minimize RCS heatup from energy added by the RCPs.

ANSWER: 073 (1.00)

b.

REFERENCE:

2-EOP-LOSC-1RCP, Rev. 4.
Salem LP LOSC02-01, EO 2; II.C & F.2, pages 8-9.
Salem LP RCPTAA-01, EO 2; II.B.2.i.2) & j, pages 9-10 & III.A.2, page 11.
Facility Bank Mod (000390).
KA [4.5/4.7]

000040K304 ..(KA's)

QUESTION: 074 (1.00)

The reactor is at 100% power and excessive steam flow has been diagnosed. Which of the following condition or conditions require initiation of a steam line isolation and entry into the EOP network (reactor trip) while performing S2.OP-AB.STM-0001(Q) "EXCESSIVE STEAM FLOW"?

1. Reactor power increasing uncontrollably.
2. RCS cooldown rate of 50 degrees F per hour.
3. Control Rods begin to step 'OUT' in fast speed.
4. The leak is determined to be inside containment.

a. 1 only
b. 1 and 2 only
c. 1, 2, and 3 only
d. 1, 2, 3, and 4.

ANSWER: 074 (1.00)

b.

REFERENCE:

S2.OP-AB.STM-0001 Rev. 2; step 3.1, page 1.
Salem LP ABSTM1-01, EO 2; III.C.1, page 6.
Facility Bank Mod (001030).
KA [4.4/4.7]

000040A204 ..(KA's)

QUESTION: 075 (1.00)

Unit 2 is operating at 90% power when condenser vacuum begins to decrease unexpectedly. A power reduction is directed in accordance with S2.OP-AB.COND-0001(Q) "LOSS OF CONDENSER VACUUM". The procedure directions limit the turbine load reduction ramp rate to 5% per minute or less.

Why should load rates not exceed 5%/min?

a. Prevents operation of the Steam Dumps.

b. Allows starting of ALL Condenser Vacuum Pumps.

c. Minimizes wear on the low pressure turbine last row blading.

d. Ensures Condensate Pump suction temperatures remain less than that required for NPSH.

ANSWER: 075 (1.00)

a.

REFERENCE:

S2.OP-AB.COND-00001(Q) Rev. 1; 3.0 NOTE, page 1.
S2.OP-AB.COND-00001(Q) TBD Rev. 1; 2.4, page 3.
Salem LP ABCOND-01, EO 3; III.C.1, page 6.
KA [3.9/4.1]

000051A202 ..(KA's)

QUESTION: 076 (1.00)

With a loss of all AC power, the MAJOR loss of reactor coolant inventory is due to:

a. opening of the PORVs from resultant high RCS pressure.

b. leakage past the RCP seals due to loss of cooling.

c. continued CVCS letdown to the VCT without charging.

d. volumetric shrinkage as the RCS is cooled by flow from the 23 AFW Pump.

ANSWER: 076 (1.00)

b.

REFERENCE:

Salem LP LOPA00-02, EO 3.a; I.B.2, pages 7-8.
KA [4.4/4.6]

000055A202 ..(KA's)

QUESTION: 077 (1.00)

Which one of the following describes equipment that is sequenced differently by the Safeguards Equipment Control (SEC) System for a LOPA if an SI signal is present compared to a situation without an SI signal present (LOPA only)?

a. 22 Auxiliary Feed Pump and 22 Chiller

b. 22 Component Cooling Pump and 22 RHR Pump

c. 24 SW Pump and 22 Auxiliary Feed Pump

d. 22 RHR Pump and 22 Chiller

ANSWER: 077 (1.00)

b.

REFERENCE:

2-EOP-LOPA-1 Rev. 2; sheet 1, Table A and Table B.
Salem LP SEC000-01, EO 4; , IV.A.1.c, page 24 and IV.B1.e, page 27 and IV.c, pages 27-28.
KA [4.1/4.5]

000055A106 ..(KA's)

QUESTION: 078 (1.00)

The following Unit 2 conditions exist:

- 50% power
- Two main feed pumps in service
- All systems in automatic
- 115 VAC vital bus 2A has just de-energized

Which ONE of the following describes the IMMEDIATE ACTIONS required by S2.OP.AB-115-0001(Q) "LOSS OF 2A 115V VITAL INSTRUMENT BUS"?

a. Take manual control of ALL Feedwater Valves (BF19s), and place the Rod Bank Selector Switch in MANUAL.

b. Take manual control of ALL Feedwater Valves (BF19s), and depress the Train 'B' "OFF & RESET BYPASS TAVG" pushbutton for Steam Dump control.

c. Trip the reactor, and start both motor-driven Auxiliary Feedwater pumps.

d. Trip the reactor, and depress the Train 'B' "OFF & RESET BYPASS TAVG" pushbutton for Steam Dump control.

ANSWER: 078 (1.00)

a.

REFERENCE:

S2.OP-AB.115-0001(Q) Rev. 5, 2.1 & 2.2, page 1.
Salem LP AB115V-01, EO 1; III.B.1, page 6.
Facility Bank Mod (001275).
KA [3.5/3.5]

000057A106 ..(KA's)

QUESTION: 079 (1.00)

In S2.OP-AB.115-0002, "Loss of 2B 115V Vital Instrument Bus", prior to Step 3.2, a caution states,

"A Reactor Trip will occur if any Steam Generator narrow range level decreases to equal to or less than 25%"

The reason for this trip is that the loss of power to the vital instrument bus:

a. did nothing to change the protection logic since a single SG level reaching less than 25% is the normal reactor trip logic.

b. caused a channel II low level signal which if combined with any second channel in that SG reaching less than 25% will cause a reactor trip.

c. caused a channel II low pressure signal which if combined with a channel III or IV low level of 25% on any SG will cause a reactor trip.

d. caused a channel II steam flow/feed flow mismatch which if combined with a channel III or IV low level of 25% on any SG will cause a reactor trip.

ANSWER: 079 (1.00)

d.

REFERENCE:

S2.OP-AB.115-0002(Q) Rev. 5: CAUTION 3.2, page 1.
Salem LP AB1152-01, EO 3; III.C.3, page 7.
RPS Logic Diagram 221056 B 9545-7; sheet 7, LOW FEEDWATER FLOW.
Facility Bank Mod (001494).
KA [3.7/3.9]

000057A203 ..(KA's)

QUESTION: 080 (1.00)

Which one of the following controls is located on the Hot Shutdown Panel 213?

a. Aux. Feedwater Storage Tank Makeup Valve OPEN/CLOSE switch.

b. Main Steam Isolation Valves OPEN/CLOSE switches.

c. Pressurizer Heater Backup Group ON/OFF switches.

d. Centrifugal Charging Pumps START/STOP switches.

ANSWER: 080 (1.00)

d.

REFERENCE:

S2.OP-AB.CR-0001(Q) Rev. 2, Step 3.6, page 2 (MSIV); Step 3.7, page 4 (Prz Heaters, AFST MU); and 3.9.A.1, page 4.
Salem LP ABCR01-01, EO 4.A; I.B, page 7.
KA [3.9/4.0]

000068K201 ..(KA's)

QUESTION: 081 (1.00)

In S2.OP-AB.CR-0002 "CONTROL ROOM EVACUATION DUE TO FIRE IN CONTROL ROOM, RELAY ROOM, OR CEILING OF 460/230V SWITCHGEAR ROOM", many actions are performed prior to leaving the Control Room. Even though the tasks may have been completed, the procedure later requires the tasks be completed AGAIN remotely, with the exception of tripping the reactor.

The MAIN reason these tasks are repeated is:

a. only ONE train of systems and components used to achieve and maintain Hot Shutdown conditions must be free of fire damage.

b. non-Safety related associated circuits can sustain fire damage that can affect exposed safe shutdown circuits.

c. the only Operator action taken credit for in the analysis before Control Room evacuation is Reactor Trip.

d. complete isolation from the control room is mandatory for ALL Safe Shutdown equipment circuitry.

ANSWER: 081 (1.00)

c.

REFERENCE:

S2.OP-AB.CR-0002(Q) Rev. 3, Technical Bases Document, 2.1, page 5.
Salem LP ABCR01-01, EO 3.B; IV.C.6, page 11.
Facility Bank Mod (001501).
KA [4.1/4.5]

000068K312 ..(KA's)

QUESTION: 082 (1.00)

Which ONE of the following statements describes the reason for stopping ALL RCPs at step 6 of EOP-FRCE-1 "RESPONSE TO EXCESSIVE CONTAINMENT PRESSURE"?

a. Prevent uncovery of the core due to separation of the steam-water mixture if the RCPs were to subsequently trip.

b. Prevent exceeding containment design pressure due to forced pumping of the RCS and ECCS inventory out the break.

c. Remove the RCP heat load from containment since the containment spray system is designed for conditions with the RCPs stopped.

d. Remove the RCPs from service for pump protection since CCW cooling flow is isolated to and from the RCPs.

ANSWER: 082 (1.00)

d.

REFERENCE:

2-EOP FRCE-1 ERG BASIS Rev. 2; Step 3-5 Basis, page 8.
Salem LP FRCE00-02, EO 3; II.C.6, page 10.
Facility Bank Mod (000186).
KA [3.8/4.2]

000069K301 ..(KA's)

QUESTION: 083 (1.00)

During response to inadequate core cooling, the Control Room operators have been unsuccessful in reducing core exit thermocouples to LESS THAN 1200 degrees F. EOP-FRCC-1 "RESPONSE TO INADEQUATE CORE COOLING" directs the operators to start RCP(s) in an attempt to reduce core exit temperature.

If RCP startup conditions CANNOT be established, which one of the following statements is applicable?

a. ALL RCPs will be started sequentially if core exit thermocouple temperatures remain above 1200 degrees F.

b. Only ONE RCP may be started, then normal startup conditions must be established prior to starting additional RCPs.

c. RCPs should be started only after permission is obtained from the Emergency Duty Officer.

d. RCPs will NOT be started without normal startup conditions because severe damage to the RCP could result in higher core temperatures.

ANSWER: 083 (1.00)

a.

REFERENCE:

2-EOP-FRCC-1 Rev. 3; Steps 29-31.1, sheet 3 L10 & A 17.
2-EOP FRCC-1 ERG BASIS Rev. 5; FRCC-1 Step 29 & 30 1, pages 53-55.
Salem LP FRCC00-02, EO 2.2; III.M & N & O, page 10.
KA [3.6/3.8]

000074K201 ..(KA's)

QUESTION: 084 (1.00)

An inadequate core cooling condition exists for Unit 2 with core exit thermocouples at 1400 degrees F and RVLIS indication at 33% and decreasing. A RCP is started in an attempt to establish temporary cooling to the core by:

a. natural single phase convection.

b. forced single phase radiation.

c. forced two phase flow.

d. natural two phase conduction.

ANSWER: 084 (1.00)

c.

REFERENCE:

2-EOP FRCC-1 ERG BASIS Rev. 5; FRCC-1 Step 30-31, pages 54.
Salem LP FRCC00-02, EO 3; III.C, page 8-9.
KA [4.5/4.9]

000074K103 ..(KA's)

QUESTION: 085 (1.00)

Unit 2 was at 50%. S2.OP-AB.RCP-0001(Q) "REACTOR COOLANT PUMP ABNORMALITY" was entered and a power decrease has initiated with all systems in automatic due to 24 RCP problems.

Current plant conditions:

Reactor Power - 34%
Pzr level - 45%
Pzr pressure - 2245 psig
SG levels at program

Assuming NO operator action, the 24 RCP trips and a reactor trip follows. What is the cause of the reactor trip?

a. Low RCS flow in ONE loop since reactor power was above the P-8 setpoint.

b. Pressurizer pressure reached the trip setpoint since spray flow was lost.

c. Shrink in the 24 SG caused level to decrease below the trip setpoint.

d. RCS heatup caused Pressurizer level to reach the trip setpoint.

ANSWER: 085 (1.00)

c.

REFERENCE:

S2.OP-AB.RCP-0001(Q) Rev. 2, ATTACHMENT 1 B.1, page 9.
Salem LP ABRCP1-01, EO 4.A; I.B, page 5.
KA [3.6/3.7]

000007A104 ..(KA's)

QUESTION: 086 (1.00)

As a MINIMUM, what is required to be checked to confirm a reactor trip upon entry into 2-EOP-TRIP-1 "REACTOR TRIP OR SAFETY INJECTION"?

a. IRPI indicate rods on bottom, and REACTOR TRIP light on RP4 lit.

b. IRPI indicate rods on bottom, and Power Range neutron flux is less than 5%.

c. Reactor trip breaker and associated bypass breaker open, and REACTOR TRIP LITE on RP4 lit.

d. Reactor trip breaker and associated bypass breaker open, and Power Range neutron flux is less than 5%.

ANSWER: 086 (1.00)

a.

REFERENCE:

Salem LP TRIP01-05, EO 3.c; III, page 7.
KA [4.3/4.5]

000007A206 ..(KA's)

QUESTION: 087 (1.00)

Which one of the following indicates that the PRT rupture disk is ruptured following a pressurizer PORV failing OPEN and NOT isolated?

a. Pressurizer level decreases.

b. PRT temperature decreases.

c. Actual PRT level decreases.

d. Relief line temperature increases.

ANSWER: 087 (1.00)

b.

REFERENCE:

Salem LP LOCATA-00, EO 3; E.3.c, page 15.
KA [3.2/3.7]

000008K101 ..(KA's)

QUESTION: 088 (1.00)

During the initial stage of a Small Break LOCA, Pressurizer level is a measure of the liquid mass in the RCS except if the break is in the vapor space of the Pressurizer vessel.

If the leak is in the Pressurizer vapor space,:

a. the level instrumentation is NOT environmentally qualified for the conditions in the Pressurizer compartment.

b. rapid evaporation causes superheat conditions in the vapor space above the liquid surface.

c. injection of much cooler ECCS water results in a large instrumentation error.

d. saturation conditions creates voids in the reactor vessel head and hot legs.

ANSWER: 088 (1.00)

d.

REFERENCE:

Transient and Accident Analysis Book, Chapter 8 SBLOCA; MODE 1 Decay Heat by Natural Circulation with the Pressurizer Controlling Pressure, page 2-11.
Salem LP LOCATA, EO 3.
KA [3.7/4.4]

000008K301 ..(KA's)

QUESTION: 089 (1.00)

In response to an RCS leak, S2.OP-AB.RC-0001(Q) "REACTOR COOLANT SYSTEM LEAK" directs the operator to transfer to ONE centrifugal charging pump in service and to reduce letdown flow to ONE 45 gpm orifice. If Pressurizer level cannot be maintained in this arrangement, then SI should be initiated.

Salem management has determined that SI actuation is required with these conditions because the leak rate at normal operating pressure is greater than approximately:

a. 45 gpm.

b. 75 gpm.

c. 100 gpm.

d. 150 gpm.

ANSWER: 089 (1.00)

c.

REFERENCE:

S2.AB.RC-0001(Q) Rev. 0 Technical Bases Document, 2.4 1st para., page 4.
Facility Bank Mod (0001036).
KA [4.2/4.5]

000009K321 ..(KA's)

QUESTION: 090 (1.00)

Following an SI actuation due to a Small Break LOCA condition, RCS pressure stabilizes at 1050 psig. Which set of parameters would be monitored to determine if an inadequate core cooling condition existed?

a. SG pressures and Core delta-T

b. RVLIS Dynamic Range and Subcooling Margin

c. RCS pressure and RCS wide range Thot temperatures

d. Core Exit Thermocouple temperatures and RVLIS Full Range

ANSWER: 090 (1.00)

d.

REFERENCE:

2-EOP-Trip-1 Rev. 4; Continuous Action Summary, sheet 2 D3.
Salem LP TRIP1-05, EO 3 & 6; VIII, page 13 & TABLE 1.
Facility Bank Mod (000113).
KA [4.3/4.7]

000009A239 ..(KA's)

QUESTION: 091 (1.00)

Unit 2 is at 100% with the following conditions:

Pressurizer level - 60% decreasing
VCT level - 55% increasing
All SEAL WATER FLOW LO bezel alarms lit.
LTDWN HX OUT TEMP HI alarm (OHA E-41) lit.

Which one of the following explains these conditions?

a. Pressurizer PORV open.

b. Letdown isolation.

c. Small break LOCA.

d. Loss of charging flow.

ANSWER: 091 (1.00)

d.

REFERENCE:

SP-206 Appendix 11, Malfunction # MTP-034, 23 CHARGING PUMP TRIP, E. General Sequence, page 3.
S2.OP-AR.ZZ-0011(Q) Rev. 0; Bezel 2-5 through 2-8 SEAL WATER FLOW LO.
S2.OP-AR.Z-0005(Q) Rev. 1: Window E-41 LTDWN HX OUT TEMP HI.
KA [3.2/3.7]

000022A202 ..(KA's)

QUESTION: 092 (1.00)

Given the following Unit 2 conditions:

Mode 5 with the 21 RHR pump in service for cooldown		
Time after reactor shutdown	-	100 hours
RCS pressure	-	300 psig
RCS Tavg	-	150 degrees F
Pressurizer level	-	10% (cold calibrate) stable
SG levels	-	70% stable for each

If the 21 RHR pump was stopped due to indications that the pump became gas bound, what is the preferred method to restore cooling to the RCS in accordance with S2.OP-AB.RHR-0001(Q) "LOSS OF RHR"?

a. Align and start the 22 RHR Pump at full flow.

b. Use reflux cooling with ALL Steam Generators.

c. Initiate Cold Leg Injection using BOTH SI Pumps.

d. Initiate Hot Leg Injection using ONE Centrifugal Charging Pump.

ANSWER: 092 (1.00)

a.

REFERENCE:

S2.OP-AB.RHR-00001(Q) Rev. 2, Technical Basis Document, 2.4 4th para., page 7.
Salem LP ABRHRO-01, EO 2; III.C, page 6.
KA [3.1/3.4]

000025K301 ..(KA's)

QUESTION: 093 (1.00)

Unit 2 is in MODE 4 with RHR system providing the shutdown cooling. All systems are in normal lineup for these conditions.

Valve 2RH25 RHR discharge relief starts to discharge (lift). Which of the following indications would you expect to receive as a result of this relief valve lifting, assuming NO operation action is taken?

a. Aux Bldg sump level increasing.

b. Console alarm "PRT LEVEL HI-LO".

c. Console alarm "VCT PRESSURE HI-LO".

d. RCDT pump runs showing on the alarm typewriter.

ANSWER: 093 (1.00)

b.

REFERENCE:

Salem LP PZRPRT-06, EO 9.e; II.G.5.e, page 18.
Facility Bank Mod (000407).
KA [3.2/3.2]

000025K202 ..(KA's)

QUESTION: 094 (1.00)

Which one of the following describes a condition for a SG Tube Leak that requires initiation of Safety Injection once the plant shutdown has been started?

a. Level difference between the affected SG and the other SGs exceeds 5%.

b. Steam flow/Feed flow mismatch exceeds 1% indicated difference for any SG.

c. 2R15 Condenser Air Ejector Monitor and any 2R19 SG Blowdown Monitor in alarm.

d. Charging Pump suction aligned to the RWST is required due to inadequate VCT level.

ANSWER: 094 (1.00)

d.

REFERENCE:

S2.OP-AB.SG-0001(Q) Rev. 1 Technical Bases Document, 2.4 4th para. (Step 3.2), page 6.
Salem LP ABSG01-01, EO 3; III.C.1, page 7.
KA [3.3/4.1]

000037A212 ..(KA's)

QUESTION: 095 (1.00)

Unit 2 has experienced a SG Tube Rupture on the 24 SG. A cooldown of the RCS has been commenced.

Given the following parameters:

RCS Tavg	-	378 degrees F
RCS pressure	-	700 psig
Przr level	-	40% increasing
Unaffected SG levels	-	30% NR stable
24 SG level	-	60% NR decreasing

What action should be taken to STABILIZE the ruptured SG level?

a. Cycle Pressurizer heaters.

b. Increase CVCS charging flow.

c. Open Pressurizer sprays.

d. Place the 75 gpm letdown orifice in service.

ANSWER: 095 (1.00)

a.

REFERENCE:

2-EOP-SGTR-2 Rev. 4; Table B, sheet 3 K20.
Salem LP SGTR02-00, EO 5.
KA [4.2/4.4]

000038A215 ..(KA's)

QUESTION: 096 (1.00)

Unit 2 is at 45% power with the 21 Feedwater Pump in service and the 22 Feedwater Pump warmed up and running at idle. What IMMEDIATE ACTION is required if the 21 Feedwater Pump trips?

a. Trip the reactor.

b. Reduce turbine load to less than 40%.

c. Open the Bypass Polisher Valves 21-23 CN108.

d. Use the SGFP Master in MANUAL to raise the 22 SGFP speed.

ANSWER: 096 (1.00)

c.

REFERENCE:

S2-OP-AB.CN0001(Q) Rev. 5; 2.2, page 1.
Salem LP ABCN01-01, EO 1; III.B.1.b, page 6.
KA [3.2/3.2]

000054G010 ..(KA's)

QUESTION: 097 (1.00)

A loss of Control Air is in progress on Unit 2. The operator is directed by S2.OP-AB.CA-0001(Q) "LOSS OF CONTROL AIR" to start the #2 Emergency Air Compressor (EAC) if 2B header pressure falls below a specific value. This pressure, which also coincides with the automatic start setpoint for the EAC, is at approximately:

a. 105 psig.

b. 90 psig.

c. 85 psig.

d. 70 psig.

ANSWER: 097 (1.00)

c.

REFERENCE:

S2.OP-AB.CA-0001(Q) Rev. 2, Step 3.5-3.7, pages 1 & 2.
Salem LP ABCA01-01, EO 4.B; II.B, page 5.
Salem LP CONAIR-07, EO 6; IV.A.1.c 2nd para., page 21.
KA [3.5/3.4]

000065A104 ..(KA's)

QUESTION: 098 (1.00)

The detector for 2R5, Spent Fuel Pool Fuel Handling Building area radiation monitor, fails causing a HIGH alarm. What actuation(s) result from this failure?

a. None, the are NO interlocks associated with this monitor.

b. All motion for the Spent Fuel Handling Crane is blocked.

c. The Fuel Handling Building ventilation is aligned to the HEPA and charcoal filters.

d. The Fuel Handling Building ventilation is aligned to the HEPA and charcoal filters, and the Control Room intake duct is isolated.

ANSWER: 098 (1.00)

c.

REFERENCE:

S2.OP-AB.RAD-0001(Q) Rev. 5; ATTACHMENT 3 1.A, page 11.
Salem LP RMS000-00, EO 2.be - e; IV.C.5, page 71.
KA [3.6/3.6]

000061A101 ..(KA's)

QUESTION: 099 (1.00)

The following Unit 2 conditions exist:

Reactor Power	-	100%
Bank D Rods(both groups)	-	215 steps in MANUAL
RCS Tavg (each loop)	-	576, 574, 577, 575 degrees F
Pressurizer pressure	-	2240 psig
Pressurizer level	-	60%
23 Charging Pump	-	in service

Which one describes the response of the Pressurizer (PZR) level controller if RCS loop 22 narrow-range Tcold instrument fails HIGH for current conditions?

a. Charging Pump speed increases since PZR program level setpoint increases to 100%.

b. Charging Pump speed decreases since actual PZR level rises when letdown isolates.

c. No effect, because PZR level controller reference setpoint will NOT change.

d. No effect, because the PZR level controller response is independent of all narrow-range RCS loop Tcold instruments.

ANSWER: 099 (1.00)

c.

REFERENCE:

Salem LP ICF000-04, EO 7.d; IV.B.2.d, page 27.
Facility Bank Mod (001247).
KA [3.4/3.8]

000028A202 ..(KA's)

QUESTION: 100 (1.00)

Unit 2 is operating with ALL SECs in Mode III following an accident. Which one of the following accurately describes when you would expect the RP4 status lights to deenergize?

a. The SI signal is reset.

b. The SEC for an associated Vital Bus is reset.

c. TWENTY minutes after the loading sequence is complete.

d. The RP1 Block Switches are taken to the Blocked position.

ANSWER: 100 (1.00)

a.

REFERENCE:

Salem LP SEC000-01, EO 2.d & 6.b; III.C.3, pages 18-19.
Facility Bank Mod (00863).
KA [3.7/3.8]

000056A238 ..(KA's)

(********** END OF EXAMINATION **********)

A N S W E R K E Y

MULTIPLE CHOICE

001	c
002	d
003	c
004	d
005	d
006	c
007	c
008	d/c
009	d
010	d
011	d
012	b
013	c
014	d
015	c
016	a
017	b
018	d
019	a
020	b
021	d
022	b
023	a
024	c
025	b
026	a
027	b
028	a
029	c
030	d
031	c
032	c
033	d
034	b
035	a
036	a
037	b
038	c
039	c
040	b
041	a
042	c
043	b
044	c
045	b

A N S W E R K E Y

046 b

MULTIPLE CHOICE

047 d
048 c
049 d
050 a
051 b
052 b
053 c
054 b
055 d
056 c
057 b
058 d
059 b
060 a
061 d
062 a
063 a
064 b
065 d
066 d
067 c
068 b
069 b
070 c
071 a
072 a
073 b
074 b
075 a
076 b
077 b
078 a
079 d
080 d
081 c
082 d
083 a
084 c
085 c
086 a
087 b
088 d
089 c
090 d

A N S W E R K E Y

091 d

MULTIPLE CHOICE

092 a

093 b

094 d

095 a

096 c

097 c

098 c

099 c

100 a

(********** END OF EXAMINATION **********)

R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
001	1.00	9000003
002	1.00	9000004
003	1.00	9000005
004	1.00	9000006
005	1.00	9000008
006	1.00	9000010
007	1.00	9000011
008	1.00	9000013
009	1.00	9000014
010	1.00	9000016
011	1.00	9000017
012	1.00	9000018
013	1.00	9000024
014	1.00	9000025
015	1.00	9000026
016	1.00	9000027
017	1.00	35915
018	1.00	9000029
019	1.00	9000030
020	1.00	9000031
021	1.00	9000032
022	1.00	36160
023	1.00	36162
024	1.00	9000035
025	1.00	9000036
026	1.00	9000037
027	1.00	9000038
028	1.00	9000039
029	1.00	9000040
030	1.00	9000041
031	1.00	9000042
032	1.00	9000043
033	1.00	9000045
034	1.00	9000046
035	1.00	9000047
036	1.00	9000048
037	1.00	9000049
038	1.00	9000050
039	1.00	9000051
040	1.00	9000052
041	1.00	9000053
042	1.00	9000054
043	1.00	9000055
044	1.00	9000056
045	1.00	9000057
046	1.00	9000058
047	1.00	9000059
048	1.00	35916
049	1.00	9000061

R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
050	1.00	9000062
051	1.00	9000063
052	1.00	9000064
053	1.00	9000065
054	1.00	35911
055	1.00	9000067
056	1.00	9000068
057	1.00	9000070
058	1.00	35917
059	1.00	9000072
060	1.00	9000073
061	1.00	9000074
062	1.00	9000075
063	1.00	9000076
064	1.00	9000078
065	1.00	9000079
066	1.00	9000080
067	1.00	9000081
068	1.00	35392
069	1.00	9000085
070	1.00	9000086
071	1.00	9000087
072	1.00	9000089
073	1.00	9000090
074	1.00	9000091
075	1.00	9000092
076	1.00	9000093
077	1.00	9000095
078	1.00	9000096
079	1.00	9000097
080	1.00	9000101
081	1.00	9000102
082	1.00	9000103
083	1.00	9000104
084	1.00	9000105
085	1.00	9000107
086	1.00	9000108
087	1.00	35374
088	1.00	9000110
089	1.00	9000111
090	1.00	9000112
091	1.00	31479
092	1.00	9000115
093	1.00	9000116
094	1.00	9000118
095	1.00	9000119
096	1.00	9000121
097	1.00	9000122
098	1.00	9000124

R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
099	1.00	9000125
100	1.00	9000127
	100.00	
	100.00	

R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

PLANT WIDE GENERICS

QUESTION	VALUE	KA
010	1.00	194001A101
002	1.00	194001A102
013	1.00	194001A103
011	1.00	194001A109
009	1.00	194001A111
008	1.00	194001A112
012	1.00	194001A116
003	1.00	194001K101
004	1.00	194001K101
005	1.00	194001K102
006	1.00	194001K103
007	1.00	194001K103
001	1.00	194001K105
PWG Total	13.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
017	1.00	001000G008
014	1.00	001000K403
016	1.00	001010A103
015	1.00	001010K404
028	1.00	003000K103
053	1.00	003000K404
031	1.00	004020A108
057	1.00	004020A305
032	1.00	004020A401
042	1.00	013000K113
054	1.00	013000K403
038	1.00	013000K411
058	1.00	015000A402
025	1.00	015000G001
023	1.00	015000K401
024	1.00	015000K402
064	1.00	015000K406
055	1.00	017000G007
027	1.00	059000A411
044	1.00	061000K402
034	1.00	061000K404
035	1.00	061000K407
056	1.00	072000G010

R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
PS-I Total	23.00	

Group II

QUESTION	VALUE	KA
030	1.00	002000G005
033	1.00	006000K410
020	1.00	010000A302
021	1.00	010000A403
029	1.00	010000K103
062	1.00	011000A101
019	1.00	012000A101
049	1.00	012000K201
026	1.00	012000K604
048	1.00	014000G005
018	1.00	016000K101
037	1.00	026000A301
036	1.00	026000G004
047	1.00	035010K503
050	1.00	062000K410
039	1.00	064000A307
052	1.00	064000G007
051	1.00	064000G010
041	1.00	064000K402
043	1.00	086000A202
PS-II Total	20.00	

Group III

QUESTION	VALUE	KA
063	1.00	005000G010
040	1.00	008000A401
059	1.00	027000A403
061	1.00	028000K601
022	1.00	041020A102
046	1.00	041020K302
045	1.00	076000A302
060	1.00	078000K303
PS-III Total	8.00	

R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

PLANT SYSTEMS

QUESTION	VALUE	KA
PS Total	51.00	

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
067	1.00	000005A101
066	1.00	000005A203
069	1.00	000015G007
070	1.00	000026G012
074	1.00	000040A204
073	1.00	000040K304
075	1.00	000051A202
077	1.00	000055A106
076	1.00	000055A202
078	1.00	000057A106
079	1.00	000057A203
080	1.00	000068K201
081	1.00	000068K312
082	1.00	000069K301
084	1.00	000074K103
083	1.00	000074K201
EPE-I Total	16.00	

Group II

QUESTION	VALUE	KA
065	1.00	000003K309
085	1.00	000007A104
086	1.00	000007A206
087	1.00	000008K101
088	1.00	000008K301
090	1.00	000009A239
089	1.00	000009K321
068	1.00	000011A210
091	1.00	000022A202
093	1.00	000025K202
092	1.00	000025K301
071	1.00	000029G010
072	1.00	000029K312
094	1.00	000037A212
095	1.00	000038A215

R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
096	1.00	000054G010
098	1.00	000061A101
EPE-II Total	17.00	

Group III

QUESTION	VALUE	KA
099	1.00	000028A202
100	1.00	000056A238
097	1.00	000065A104
EPE-III Total	3.00	
EPE Total	36.00	
Test Total	100.00	

ATTACHMENT 2

SRO EXAM AND MASTER EXAMINATION ANSWER KEY

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 1

CANDIDATE'S NAME: _____

FACILITY: Salem 1 & 2

REACTOR TYPE: PWR-WEC4

DATE ADMINISTERED: 94/06/20

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%
100.00	_____	___ % TOTALS

FINAL GRADE

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

_____
Candidate's Signature

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

MULTIPLE CHOICE

001	a	b	c	d	___
002	a	b	c	d	___
003	a	b	c	d	___
004	a	b	c	d	___
005	a	b	c	d	___
006	a	b	c	d	___
007	a	b	c	d	___
008	a	b	c	d	___
009	a	b	c	d	___
010	a	b	c	d	___
011	a	b	c	d	___
012	a	b	c	d	___
013	a	b	c	d	___
014	a	b	c	d	___
015	a	b	c	d	___
016	a	b	c	d	___
017	a	b	c	d	___
018	a	b	c	d	___
019	a	b	c	d	___
020	a	b	c	d	___
021	a	b	c	d	___
022	a	b	c	d	___
023	a	b	c	d	___
024	a	b	c	d	___
025	a	b	c	d	___
026	a	b	c	d	___
027	a	b	c	d	___
028	a	b	c	d	___
029	a	b	c	d	___
030	a	b	c	d	___
031	a	b	c	d	___
032	a	b	c	d	___
033	a	b	c	d	___
034	a	b	c	d	___
035	a	b	c	d	___
036	a	b	c	d	___
037	a	b	c	d	___
038	a	b	c	d	___
039	a	b	c	d	___
040	a	b	c	d	___
041	a	b	c	d	___
042	a	b	c	d	___
043	a	b	c	d	___
044	a	b	c	d	___
045	a	b	c	d	___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

046 a b c d ___

MULTIPLE CHOICE

047 a b c d ___
048 a b c d ___
049 a b c d ___
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082 a b c d ___
083 a b c d ___
084 a b c d ___
085 a b c d ___
086 a b c d ___
087 a b c d ___
088 a b c d ___
089 a b c d ___
090 a b c d ___

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

If you change your answer, write your selection in the blank.

091 a b c d ___

MULTIPLE CHOICE

092 a b c d ___

093 a b c d ___

094 a b c d ___

095 a b c d ___

096 a b c d ___

097 a b c d ___

098 a b c d ___

099 a b c d ___

100 a b c d ___

(********** END OF EXAMINATION **********)

QUESTION: 001 (1.00)

Which of the following individuals is responsible for daily visual accountability of the security keys on the Unit 1 Primary Operator key ring?

a. Primary Operator

b. Shift Clerk

c. Nuclear Control Operator

d. Control Room Nuclear Shift Supervisor

ANSWER: 001 (1.00)

c.

REFERENCE:

Key Control, AD-37, Rev. 11, page 4.
Facility Learning Objective not identified.

[3.1/3.4*]

194001K105 ..(KA's)

QUESTION: 002 (1.00)

A locked throttle valve is include on a valve lineup that is to be independently verified.

Select the required action for performing the lineup and independent verification for this valve.

a. The initial verifier and independent verifier will simultaneously verify the lock installed and that the valve appears to be in the correct position.

b. The initial verifier will unlock the valve, fully open the valve, reclose it the required number of turns, then relock the valve. The independent verifier will verify the valve correctly locked.

c. The initial verifier will unlock the valve, fully close the valve, reopen it the required number of turns, then relock the valve. The independent verifier will observe the actions of the initial verifier.

d. The initial verifier will unlock the valve, fully close the valve, reopen it the required number of turns, then relock the valve. The independent verifier will use other methods to verify position where possible.

ANSWER: 002 (1.00)

c.

REFERENCE:

System Alignments, SC.OP-DD.ZZ-OD7, Rev 0, page 3 and 4.
Station Operating Practices, AR1A00-09, Learning Objective 4.

[3.6/3.7]

194001K101 ..(KA's)

QUESTION: 003 (1.00)

As second verifier, you identify a valve that is closed, however, its required to be in the open position.

Select the required action.

a. Open the valve. Note on the valve lineup that the valve was found in the open position.

b. Contact the initial verifier. After the initial verifier positions the valve perform the independent verification.

c. Open the valve. Notify the Nuclear Shift Supervisor.

d. Immediately contact the Nuclear Shift Supervisor for resolution.

ANSWER: 003 (1.00)

d.

REFERENCE:

System Alignments, SC.OP-DD.ZZ-OD7, Rev 0, page .
Station Operating Practices, AR1A00-09, Learning Objective 4.

[3.6/3.7]

194001K101 ..(KA's)

QUESTION: 004 (1.00)

A jumper has been installed as a temporary modification per an approved troubleshooting package.

When is a temporary modification package (TMP) required to be prepared for this jumper?

a. Prior to initiating troubleshooting.

b. Prior to any shift change with the jumper installed.

c. Anytime the troubleshooting is to be stopped.

d. If troubleshooting is to be stopped for more than 16 hours.

ANSWER: 004 (1.00)

d.

REFERENCE:

Control of Temporary Modifications, NC.NA-AP.ZZ-0013, Rev 1, Page 9.
Miscellaneous Administrative Procedures, MSCAP-01, Learning Objective 2.

[3.1/4.1*]

194001A112 ..(KA's)

QUESTION: 005 (1.00)

The person named on a tag is not on site and cannot be located. It is required that the tag be cleared to support plant operations. Work has been completed on the component.

Select the individual who is authorized to sign the Tagging Release or direct the Nuclear Shift Supervisor to sign the release.

a. Operations Manager

b. The individual's Department Manager.

c. The job supervisor.

d. Senior Nuclear Shift Supervisor.

ANSWER: 005 (1.00)

a.

REFERENCE:

Safety Tagging Program, NC.NA-AP.ZZ-0015, Rev 1, Page 10.
Station Safety Tagging, AR30000-04, Learning Objective 5.

[3.7/4.1]

194001K102 ..(KA's)

QUESTION: 006 (1.00)

What is the administrative annual limit on Total Effective Dose Equivalent (TEDE) for category 2 workers at Salem?

a. 0.5 rem

b. 1.0 rem

c. 2.0 rem

d. 3.0 rem

ANSWER: 006 (1.00)

c.

REFERENCE:

Radiation Protection Program, NC.NA-AP.ZZ-0024, Rev 3, Attachment 2.
Radiological Control Program, RPP000-05, Learning Objective 2.

[2.8/3.4]

194001K103 ..(KA's)

QUESTION: 007 (1.00)

Which of the following would be the maximum dose rate that would NOT require a door to be locked?

a. 99 mrem/hr

b. 499 mrem/hr

c. 999 mrem/hr

d. 1999 mrem/hr

ANSWER: 007 (1.00)

c.

REFERENCE:

Radiation Protection Program, NC.NA-AP.ZZ-0024, Rev 3, Attachment 2.
Radiological Control Program, RPP000-05, Learning Objective 3.

[2.8/3.4]

194001K103 ..(KA's)

QUESTION: 008 (1.00)

An operator was on jury duty from March 1 to June 22. The operator's license was in an active status prior to the jury duty.

Select the MINIMUM number of shifts that the operator will have to stand and the date by which those watches must be completed in order to maintain his license active.

a. The operator will have to stand 3 twelve hour shifts prior to July 1.

b. The operator will have to stand 5 twelve hour shifts prior to July 1.

c. A 25% grace period is allowed by Technical Specifications so the operator will have until July 22 to complete 3 twelve hour shifts.

d. A 25% grace period is allowed by Technical Specifications so the operator will have until July 22 to complete 5 twelve hour shifts.

ANSWER: 008 (1.00)

b.

REFERENCE:

Station Operating Procedures, NC.NA-AP.ZZ-0005, Rev. 5, page 11.
Station Operating Practices, AR1A00-09, Learning Objective 5.

[2.5/3.4]

194001A103 ..(KA's)

QUESTION: 009 (1.00)

An instrument technician is required to enter a cabinet that is posted "Trip Hazard."

Which of the following identifies the restrictions associated with entry into this cabinet?

a. The posting is only to warn the technician to exercise extra care. No specific permission for entry is required.

b. The technician is authorized entry verbally by the Nuclear Control Operator (NCO).

c. The technician is authorized entry verbally by the Nuclear Shift Supervisor.

d. The technician is authorized entry into the cabinet by the signed authorization of a work order.

ANSWER: 009 (1.00)

d./c.

REFERENCE:

Station Operating Procedures, NC.NA-AP.ZZ-0005, Rev. 5, page 11.
Facility Learning Objective Not Identified.

[3.1/4.1*]

194001A112 ..(KA's)

QUESTION: 010 (1.00)

A reactor operator trainee is assigned to your shift. A reactor startup is to be performed.

Who is required to grant permission for the trainee to withdraw the control rods to perform the startup?

a. The trainee is granted permission by being enrolled in the licensing class.

b. The NSS must grant permission for the trainee to perform the startup.

c. The SNSS must grant permission and obtain verbal concurrence from the Operations Manager.

d. The Operations Manager must issue written authorization.

ANSWER: 010 (1.00)

d.

REFERENCE:

Station Operating Procedures, NC.NA-AP.ZZ-0005, Rev. 5, page 28.
Station Operating Practices, AR1A00-09, Learning Objective 3.

[2.8/4.1*]

194001A111 ..(KA's)

QUESTION: 011 (1.00)

A Limiting Condition for Operation (LCO) that has a requirement to be in HOT SHUTDOWN within 12 hours, has just been entered. The problem is expected to be resolved in a timely manner.

In order to proceed to a Hot Shutdown in a controlled manner, what is the MAXIMUM hours that can elapse before the shutdown is required to be commenced?

a. 1 hour

b. 3 hours

c. 6 hours

d. 11 hours

ANSWER: 011 (1.00)

c.

REFERENCE:

Station Operating Procedures, NC.NA-AP.ZZ-0005, Rev. 5, page 38.
Facility Learning Objective Not Identified.

[2.5/3.4]

194001A103 ..(KA's)

QUESTION: 012 (1.00)

The plant was operating at 45% when the feedwater regulating valve (BF-19) for 12 Steam Generator was placed in manual due to oscillations in the controller.

What is the requirement for operation with the valve in manual?

a. The Board NCO shall remain in the vicinity of the control [illegible] and only perform actions that will not interfere with monitoring the steam generator.

b. The Board NCO is required to announce that a valve has been placed in manual so that all control room personnel can provide additional monitoring of steam generator level.

c. An NCO, other than the Board NCO, is required to be assigned to monitor steam generator level. This individual can perform other tasks that do not interfere with monitoring of steam generator level.

d. An NCO, other than the Board NCO, is required to be continuously stationed at the BF-19 controller. The NCO can have no other responsibilities.

ANSWER: 012 (1.00)

d.

REFERENCE:

Salem Operations Standing Orders, item Q, page 13.
Facility Learning Objective not Identified.

[2.7/3.9*]

194001A109 ..(KA's)

QUESTION: 013 (1.00)

Select under what conditions, if any, the Emergency Coordinator can downgrade a protective action recommendation(PAR)?

a. If the initial PAR was based on incorrect data.

b. When the emergency is downgraded from a General Emergency.

c. When the release is terminated.

d. The Emergency Coordinator cannot downgrade a PAR.

ANSWER: 013 (1.00)

d.

REFERENCE:

Licensed Operators/Emergency Response, 0215.000.00B-00002-02, page 16, LO 7.

[3.1/4.4]

194001A116 ..(KA's)

QUESTION: 014 (1.00)

Select the emergency classification(s) when accountability is to be performed.

a. Required at Site Area Emergency and General Emergency for all classifications.
Required at Alert only for radiological releases.

b. Required at General Emergency.
May be implemented at Site Area Emergency or Alert.

c. Required at Site Area Emergency and General Emergency.
May be implemented at Alert or Unusual Event.

d. Required at Alert or higher.
May be implemented at Unusual Event.

ANSWER: 014 (1.00)

c.

REFERENCE:

Licensed Operators/Emergency Response, 0215.000.00B-00002-02, page 16, LO 7.

[3.1/4.4]

194001A116 ..(KA's)

QUESTION: 015 (1.00)

Salem Station is in an ALERT condition. How often is the Station Status Checklist (SSCL) retransmitted to the designated agencies after the initial transmittal has been made.

a. Every 15 minutes

b. Every 30 minutes

c. Every 45 minutes

d. Every 60 minutes

ANSWER: 015 (1.00)

b.

REFERENCE:

Facility Exam Bank 000715
Licensed Operators/Emergency Response, 0215.000.00B-00002-02, page 32, LO 13.

[3.1/4.4]

194001A116 ..(KA's)

QUESTION: 016 (1.00)

What position's turnover responsibilities include reviewing and accounting for all keys in the key locker.

a. Oncoming NSSW

b. Offgoing NSSW

c. Oncoming NSS

d. Offgoing NSS

ANSWER: 016 (1.00)

b.

REFERENCE:

Facility Exam Bank 000291
Shift Turnover Responsibilities, SC.OP-DD.ZZ-OD20, page 9, Rev. 1.
Shift Turnover and Log Keeping, AR7000-02, Learning Objective 1.

[2.5/3.4]

194001A103 ..(KA's)

QUESTION: 017 (1.00)

Maintenance on a pump has been completed. The Electrical Supervisor wishes to check proper rotation of the motor, but the motor breaker has a Red Blocking Tag installed on it.

Which of the following should be performed to allow for this testing?

a. Permanent release of tagout

b. Partial release of only the tag in question (motor breaker)

c. Temporary release of the motor breaker tag

d. Replacement of the Red Blocking Tag with a Worker's Blocking Tag on the motor breaker.

ANSWER: 017 (1.00)

c.

REFERENCE:

Facility Exam Bank 000298
Safety Tagging Program, NC.NA-AP.ZZ-0015, Rev 1, page 32.
Station Safety Tagging, AR30000-04, Learning Objective 6.

[3.7/4.1]

194001K102 ..(KA's)

QUESTION: 018 (1.00)

Control rods are in automatic. Compare the output signal from the reactor control unit on a 2% step change in turbine load with the power at 90% with a 2% step change at 40% power.

At 90% power the signal would be:

a. the same as at 40% power.

b. larger due to the response of the Non-Linear gain unit.

c. smaller due to the response of the Non-Linear gain unit.

d. smaller due to the response of the Variable gain unit.

ANSWER: 018 (1.00)

d.

REFERENCE:

Rod Control System Description, 300S-Misc/2:25, Page 14.
Rod Control System Lesson, ROC000-01, LO 2.e.

[3.5/3.8]

001000K403 ..(KA's)

QUESTION: 019 (1.00)

Which of the following instrument failures would result in a lowering in the Rod Insertion Limit?

a. The highest reading Thot fails downscale.

b. The highest reading Thot fails upscale.

c. The lowest reading Tavg output fails upscale.

d. The highest reading Tcold fails downscale.

ANSWER: 019 (1.00)

a.

REFERENCE:

Rod Position Indication System Lesson Plan, 302/304-145.12-RPI-04, page 13, LO 1/2.4.

[4.2/4.4]

001010A103 ..(KA's)

QUESTION: 020 (1.00)

Control rods are in automatic and inserting due to an instrument failure. The Rod Bank LoLo Limit (OHA E-16) is received.

This indicates that the rods have reached the Rod Insertion Limit and:

a. the rods will stop and must be reset to allow further insertion.

b. the rods will continue to insert.

c. the rods will stop but will start rod insertion again when a lower rod insertion limit is computed.

d. the rods will insert only in manual control.

ANSWER: 020 (1.00)

b.

REFERENCE:

Rod Position Indication System Lesson Plan, 302/304-145.12-RPI-04, page 13, LO 1/2.6.

[3.6/3.6]

001000G008 ..(KA's)

QUESTION: 021 (1.00)

Following a Loss of Coolant Accident the Upper Range of Reactor Vessel Level Instrument System (RVLIS) is being used to monitor level. No RCPs are running

Select the expected response of Train A and Train B of Upper Range RVLIS indications on the RVLIS summary page if #22 RCP is started.

a. Both will indicate the same as before the pump is started.

b. Train A will indicate the same as before the pump is started but Train B will indicate downscale.

c. Train A will indicate downscale but Train B will indicate the same as before the pump is started.

d. Both will indicate downscale.

ANSWER: 021 (1.00)

d.

REFERENCE:

Reactor Vessel Level Instrumentation System Description, Volume 2, Chapter 24, page 34.
Reactor Vessel Level Instrumentation System Lesson Plan, RVLISO-06, Volume 8, Chapter 28, Learning Objective 7.a.

[3.4*/3.4*]

016000K101 ..(KA's)

QUESTION: 022 (1.00)

With operation at 63% power the controlling channel I, of pressurizer pressure fails downscale. With no operator action what will be the expected response?

a. A reactor scram and safety injection.

b. Pressure cycling from 2315 to 2335 as PR1 opens and closes.

c. Pressure cycling from 2315 to 2335 as PR2 opens and closes.

d. Pressure cycling from 2315 to 2335 as both PR1 and PR2 open and close.

ANSWER: 022 (1.00)

b.

REFERENCE:

Pressurizer Pressure and Level Control System Description, Volume 2, Chapter 25, page 39.
Pressurizer Pressure and Level Control System Lesson Plan, PXRP/L-00, Volume 7, Chapter 5, Learning Objective 2.h.

[3.6/3.5]

010000A302 ..(KA's)

QUESTION: 023 (1.00)

The reactor is shutdown and a cooldown is in progress with wide range RCS temperature at 300 F. The control bezel for PORV PR1 is selected to MANUAL. STOP VALVE PR6 is closed.

Select the response of PORV PR1 and STOP VALVE PR6 if the operator places the keylock switch for channel I of Pressurizer Overpressure Protection System (POPS) to ON.

a. PR1 will remain in MANUAL. POPS will not be enabled until the operator places PR1 in AUTOMATIC. STOP VALVE PR6 will remain closed.

b. PR1 will remain in MANUAL. POPS will NOT be enabled. STOP VALVE PR6 will open.

c. PR1 will transfer to automatic mode and POPS will be enabled. STOP VALVE PR6 will remain closed.

d. PR1 will transfer to automatic mode and POPS will be enabled. STOP VALVE PR6 will open.

ANSWER: 023 (1.00)

d.

REFERENCE:

Pressurizer Pressure and Level Control System Description, Volume 2, Chapter 25, page 17.
Pressurizer Pressure and Level Control System Lesson Plan, PZRP/L-00, Volume 7, Chapter 5, Learning Objective 2.f.

[4.0/3.8]

010000A403 ..(KA's)

QUESTION: 024 (1.00)

The following Plant conditions exist:

MODE 1 with reactor power at 19%.
Tave at 555 degrees F.
Turbine Power is 16%
Steam Dumps in AUTOMATIC in MS PRESSURE CONTROL mode.

Which ONE of the following statements describes the plant response if PT-507, Steam Header Pressure, fails HIGH?

a. Steam dumps remain closed because the arming signal is absent since the difference between reactor and turbine power is 3%.

b. Steam dumps open with arming signal present since the steam header pressure is greater than the controller setpoint.

c. Steam dumps open with arming signal present since Tave-Tref has greater than a 2 degree difference.

d. Steam dumps remain closed because the arming signal is absent since Tave is less than the LOW Tave setpoint.

ANSWER: 024 (1.00)

b.

REFERENCE:

Steam Dump TSD System Description, Volume 2, Chapter 26, page 13.
Steam Dump Lesson Plan, STDUMP-03, Volume 7, Chapter 6, Learning Objective 2.f.

[3.1/3.2]

041020A102 ..(KA's)

QUESTION: 025 (1.00)

During Mode 5 operations, the operators receive the following alarms/indications

- Audible count rate increase
- High Flux at Shutdown alarm
- Source Range indicators increasing

Select the ONE response below which describes the proper actions to be taken for the given conditions.

a. Immediately open the trip breakers and initiate boration

b. Make a PA announcement to all personnel to evacuate containment, calculate SDM.

c. Place both Primary Water pumps to off and verify all dilution paths are isolated.

d. Verify both Primary Water pumps are off, close CV179, and rapid borate.

ANSWER: 025 (1.00)

b.

REFERENCE:

S2.OP-AR.ZZ-0005
TSD Chapter 20.
Facility Exam Bank 000056

[3.8/3.8]

015000G001 ..(KA's)

QUESTION: 026 (1.00)

With one bypass breaker closed, racking the second bypass breaker in will result in:

a. opening all trip and bypass breakers due to a General Warning Reactor Trip.

b. opening only the closed bypass breaker.

c. opening the trip breaker in parallel with the closed bypass breaker only.

d. a General Warning Alarm in the control room but not open any trip or bypass breakers.

ANSWER: 026 (1.00)

a.

REFERENCE:

Reactor Protection System Description, Volume 3, Chapter 28, page 17.
Reactor Protection System Lesson Plan, RXPROT-08, Volume *, Chapter 2, Learning Objective 3.

[3.3/3.6]

012000K604 ..(KA's)

QUESTION: 027 (1.00)

Which of the following events can the operator use the MANUAL RESET pushbuttons to restore feedwater capability to the steam generator?

a. A high steam line flow causes a low Tavg and an SI.

b. A steam dump fails open on a turbine trip from 65% power causing a low Tavg and reactor trip.

c. The operator overfeeds a single steam generator to the High-High Level.

d. A spray valve fails open causing RCS pressure to decrease to 1725 psig.

ANSWER: 027 (1.00)

b.

REFERENCE:

Condensate and Feedwater System Description, Volume 3, Chapter 34, page 20.
Condensate and Feedwater System Lesson Plan, CN/BFO-05, Volume 3, Chapter 18, Learning Objective 2.f.

[3.1/3.3]

059000A411 ..(KA's)

QUESTION: 028 (1.00)

A spurious Phase "A" signal has occurred.

Identify the effect this will have on RCP operation.

a. The RCP seal leakoff will be isolated. The seals will be directed to the PRT via a relief valve.

b. The CCW t RCP thermal barrier will be lost so seal injection be maintained.

c. The CCW to the RCP motor will be lost so the RCP's will have to be tripped.

d. The RCP seal injecti is isolated, so CCW flow to the thermal barrier must be ver

ANSWER: 028 (1.00)

a.

REFERENCE:

Reactor Coolant Pump System cription, Volume 1, Chapter 4, page 16.
Reactor Coolant Pump Lesson an, RCPUMP-04, Volume 1, Chapter 6, Learning Objective 15.

[3.3/3.6]

003000K103 ..(KA's)

QUESTION: 029 (1.00)

The reactor is shutdown with all reactor coolant pumps stopped. A steam bubble exists in the pressurizer. It is desired to start a RCP.

Which RCP would be most effective in providing pressure control?

a. 21

b. 22

c. 23

d. 24

ANSWER: 029 (1.00)

c.

REFERENCE:

Reactor Coolant Pump System Description, Volume 1, Chapter 4, page 18
Reactor Coolant Pump Lesson Plan, RCPUMP-04, Volume 1, Chapter 6, Learning Objective 13.

[3.6/3.7]

010000K103 ..(KA's)

QUESTION: 030 (1.00)

Leakage into #13 steam generator is determined to be 0.5 gpm. No leakage is detectable into the other steam generators. Other leakage which cannot be identified is determined to be 0.6 gpm. Leakage from known sources other than steam generator leakage is determined to be 4.0 gpm.

With these conditions in existence, Technical Specification leakage limits:

a. are not exceeded.

b. are exceeded due to the total leakage into the steam generator and unidentified leakage exceeding 1 gpm.

c. are exceed due to steam generator leakage exceeding limits for pressure boundary leakage.

d. are exceeded due to excessive leakage into one steam generator.

ANSWER: 030 (1.00)

d.

REFERENCE:

Salem Unit #1 Technical Specifications, Section 3.4.6.2.
Steam Generator Lesson Plan, STMGEN-02, Volume 1, Chapter 7, Learning Objective 13.

[3.6/4.1]

002000G005 ..(KA's)

QUESTION: 031 (1.00)

The Unit 2 CVCS Makeup Mode Selector switch is selected to AUTOMATIC.

Identify the response if VCT level decreases to the setpoint for automatic makeup.

a. Boron flow rate is set at a specific rate by the operator and Primary water flow rate is set at 70 gpm. Flow is via CV-181 to the VCT.

b. Boron flow rate is set at a 40 gpm and the primary water flow rate is set at a specific rate by the operator. Flow is via CV-181 to the VCT.

c. Boron flow rate is set at a specific rate by the operator and Primary water flow rate is set at 70 gpm. Flow is via CV-185 to the Charging Pumps suction.

d. Boron flow rate is set at a 40 gpm and the primary water flow rate is set at a specific rate by the operator. Flow is via CV-185 to the Charging Pumps suction.

ANSWER: 031 (1.00)

c.

REFERENCE:

Chemical and Volume Control System Description, Volume 1, Chapter 6, page 36.
Chemical and Volume Control System Lesson Plan, CVCS00-03, Volume 1, Chapter 8, Learning Objective 2.e.

[3.8/3.3]

004020A401 ..(KA's)

QUESTION: 032 (1.00)

A loss of coolant accident has occurred on Unit 2 that resulted in an SI. The SI signal has NOT been reset. RWST level has decreased to 15.24 ft.

Which of the following would have to be performed manually following the automatic opening of the RHR suction valves to the containment sump?

a. Closed cooling water would have to be aligned to the RHR heat exchangers.

b. The RHR discharge valve to the suction of the SI pumps (21SJ45) would have to be opened.

c. The charging pump suction cross tie valve (21SJ113) would have to opened.

d. RHR pump suctions from the RWST would have to be closed.

ANSWER: 032 (1.00)

b.

REFERENCE:

Residual Heat Removal System Description, Volume 1, Chapter 8, page 14.
Residual Heat Removal Lesson Plan, 06, Volume 1, Chapter 9, Learning Objective 2.f.

[3.1*/3.5*]

005000K408 ..(KA's)

QUESTION: 033 (1.00)

Select the action that prevents discharging water from the containment sump to the RWST via the SI pump mini-flow lines on Unit 1.

a. Mini-flow valves SJ67 and SJ68 close when SI pumps suction SJ45 is opened.

b. The power lockout switch must be placed to VALVE OPERABLE position which causes the mini-flow valves to close.

c. SJ67 and SJ68 valve control must be placed to RECIRC OVERRIDE CLOSE which causes the mini-flow valves to be overridden closed.

d. SI pump suction SJ45 is interlocked closed until either mini-flow valves is closed.

ANSWER: 033 (1.00)

d.

REFERENCE:

Emergency Core Cooling System Description, Volume 1, Chapter 10, page 16.
Emergency Core Cooling Lesson Plan, ECCS00-06, Volume 1, Chapter 10, Learn ng Objective 7.
NOTE: Facility needs to verify correct valve terminology for unit 1 valves.
[3.6*/3.7]

006000K410 ..(KA's)

QUESTION: 034 (1.00)

The AFW pump low suction pressure trips are made operable during which of the following conditions.

a. Severe weather.

b. Local Operation of the AFW pumps.

c. Alignment of AFW to an alternate suction source.

d. Potential exists for freezing of the AFWST.

ANSWER: 034 (1.00)

a.

REFERENCE:

Auxiliary Feedwater System Description, Volume 1, Chapter 11, table 2.
Auxiliary Feedwater System Lesson Plan, AFEEDW-10, Volume 1, Chapter 14, Learning Objective 4.a.

[3.1*/3.3*]

061000K407 ..(KA's)

QUESTION: 035 (1.00)

During a loss of coolant accident the containment spray system initiates but fails to inject NaOH.

Select the expected condition resulting from the failure to inject NaOH.

a. Containment sump will be acidic and increased potential for higher dose during a release from iodine.

b. Containment sump will be acidic and increased potential for hydrogen explosion.

c. Containment sump will be basic and increased potential for higher dose during a release from iodine.

d. Containment sump will be basic and increased potential for hydrogen explosion.

ANSWER: 035 (1.00)

a.

REFERENCE:

Containment and Support Systems System Description, Volume 1, Chapter 12, page 17.
Containment Spray Lesson Plan, 300S-000.0S-CSPRAY-06, Volume 1, Chapter 11, Learning Objective 1.

[3.6/3.9]

026000G004 ..(KA's)

QUESTION: 036 (1.00)

A steam break occurred inside containment concurrent with an undervoltage condition on the 4KV vital busses. Prior to the containment pressure reaching the high high containment pressure setpoint, SEC is reset but SI is NOT reset.

Based on the above conditions, select the response of the containment spray system when containment pressure reaches the high high containment pressure setpoint.

a. No action will occur.

b. The containment spray valves will reposition but the pumps will NOT start.

c. The containment spray valves will NOT reposition but the pumps will start.

d. The containment spray valves will reposition and the pumps will start.

ANSWER: 036 (1.00)

b.

REFERENCE:

Containment Spray System Description, Volume 1, Chapter 12A, page 8.
Containment Spray System Lesson Plan, Volume 1, Chapter 11, Learning Objective 2.f.

[4.3/4.5]

026000A301 ..(KA's)

QUESTION: 037 (1.00)

A Safety Injection has occurred. During bus loading an undervoltage condition occurs simultaneously on the "2A" and "2B" vital busses.

Select the mode response of SEC for all vital busses.

a. Busses "2A" and "2B" will go through Mode III sequencing; Bus "2C" will go through Mode I sequencing.

b. Busses "2A" and "2B" will go through Mode II sequencing; Bus "2C" will go through Mode I sequencing.

c. All busses will go through a Mode III sequencing.

d. All busses will go through a Mode II sequencing.

ANSWER: 037 (1.00)

c.

REFERENCE:

Safeguards Equipment Control System Description, Volume 1, Chapter 13, page 5.
Safeguards Equipment Control Lesson Plan, SEC000-01, Volume 6, Chapter 19, Learning Objective 2. Page 32.

[3.2/3.8]

013000K411 ..(KA's)

QUESTION: 038 (1.00)

Following a Mode II operation of Safeguards Equipment Control (SEC), when will the 230 V Control Centers lockouts automatically reset?

a. 5 minutes.

b. 10 minutes.

c. 20 minutes.

d. 30 minutes.

ANSWER: 038 (1.00)

c.

REFERENCE:

Safeguards Equipment Control Lesson Plan, SEC000-01, Volume 6, Chapter 19, page 15, Learning Objective 2.f.

[3.6*/3.7*]

064000A307 ..(KA's)

QUESTION: 039 (1.00)

Which of the following Control Area Ventilation operational condition lineups occurs automatically?

a. Fire Inside Control Area.

b. Fire Outside Relay Area.

c. Accident-Inside Air

d. Accident-Mixed Air

ANSWER: 039 (1.00)

c.

REFERENCE:

Control Area Ventilation System Description, Volume 4, Chapter 48, Table II.
Control Area Ventilation Lesson Plan, CAVENT-02, Volume 2, Chapter 11, Learning Objective 7.

[2.8/3.1]

013000K113 ..(KA's)

QUESTION: 040 (1.00)

Which one of the following conditions will cause the automatic start of both diesel drive fire pumps?

a. Fire main pressure decreases to 80 psig.

b. Loss of normal AC power.

c. The jockey pump trips.

d. Fire main pressure decreasing at a rate faster than 10 psi/sec.

ANSWER: 040 (1.00)

b.

REFERENCE:

Fire Protection System Description, Volume 4, Chapter 50, page 30.
Fire Protection Lesson Plan, FIRPRO-03, Volume 4, Chapter 30, Learning Objective 2.e.

[3.0/3.3]

086000A202 ..(KA's)

QUESTION: 041 (1.00)

With the plant at full power, a reactor trip signal is generated through the Solid State Protection System (SSPS) due to a Instrumentation Technician error. On the trip, the Train A Reactor Trip Breaker failed to open. Assuming no operator action is taken, select the consequence of this breaker failure.

a. No adverse consequences to the steam dump system. A single failure of an arming signal input has no affect.

b. RCS temperature will be driven to 547F via the Plant Trip Controller.

c. RCS temperature will be driven to 552F via the Load Rejection Controller instead of the Plant Trip Controller.

d. Steam dumps will not arm and RCS temperature will be maintained via the MS10's and SG safeties.

ANSWER: 041 (1.00)

b.

REFERENCE:

Facility Question Bank 000582.
Steam Dump System Description, Volume 2, Chapter 26, page 20.
Steam Dump Lesson Plan, STDUMP-03, Volume 7, Chapter 6, Learning Objective 2.

[3.8/3.9]

041020K302 ..(KA's)

QUESTION: 042 (1.00)

Given the following:

- Reactor power is at 25%.
- All plant controls are in automatic.
- Loop 21 MSIV inadvertently shuts.

WHICH ONE (1) of the following parameters would show an INITIAL INCREASE following this event? (Assume NO operator actions are taken).

a. Steam generator #21 level.

b. Steam Generator #22 pressure.

c. Loop #24 Cold leg temperature.

d. Steam Generator #23 level.

ANSWER: 042 (1.00)

d.

REFERENCE:

Facility Question 001346.
Steam Generator Water Level Control System Description, Volume 2, Chapter 27, page 7.
Steam Generator Water Level Control System Lesson Plan, Volume 7, Chapter 08, Learning Objective 8.

[2.8/3.1]

035010K503 ..(KA's)

QUESTION: 043 (1.00)

The group demand counters for Control Bank D are at 100 steps.

In order to satisfy Technical Specification limits, individual rod positions for rods in Control Bank D must indicate from:

a. 88 to 100 steps.

b. 100 to 112 steps.

c. 88 to 112 steps.

d. 96 to 104 steps.

ANSWER: 043 (1.00)

c.

REFERENCE:

Rod Position Indication System Lesson Plan, 302S-RPI-04, Volume 7, Chapter 4, Learning Objective 3.0.
Technical Specifications 3.1.3.1.

[3.1/3.7]

014000G005 ..(KA's)

QUESTION: 044 (1.00)

The Unit is at full power when the 115 VAC infeed breaker to the Output Bay of SSPS train 'A' inadvertently opens.

This event will:

a. cause a reactor trip.

b. cause an SI actuation.

c. have no effect since this power supply is auctioneered with 115 VAC from the Logic Bay.

d. prevent an SI actuation from Train 'A'.

ANSWER: 044 (1.00)

d.

REFERENCE:

Facility Question Bank 000772
Reactor Protection System Description, Volume 3, Chapter 28, page 21.
Reactor Protection System Lesson Plan, RXPROT-08, Volume 8, Chapter 2, Learning Objective 2.g.

[3.3/3.7]

012000K201 ..(KA's)

QUESTION: 045 (1.00)

Select the mechanism used by the 115 V Vital Power to transfer power to the 125 VDC power supply.

a. A power auctioneering circuit transfers power from the Normal AC Input to the 125 V DC supply.

b. A static switch transfers power from 230 V AC emergency power to 125 V DC supply.

c. When low voltage is sensed from the 230 V AC Emergency Power the 125 V DC input breaker is automatically shut.

d. A power auctioneering circuit transfers power from the 230 V DC Emergency Power to 125 V DC.

ANSWER: 045 (1.00)

a.

REFERENCE:

Electrical Distribution System Description, Volume 4, Chapter 42D, page 2.
115 VAC Control Power System Lesson Plan, Volume 5, Chapter 08, Learning Objective 2.e.

[3.1/3.5]

062000K410 ..(KA's)

QUESTION: 046 (1.00)

A loss of off-site power has occurred. The operator observes that 2A Diesel is operating at 2950 KW.

Select the MINIMUM required action for continued operation of 2A diesel generator.

a. Reduce load to less than 2860 immediately.

b. Reduce load to less than 2860 within 30 minutes.

c. Reduce load to less than 2360 within 2 hours.

d. Reduce load to less than 2750 within 30 minutes.

ANSWER: 046 (1.00)

b.

REFERENCE:

EDG & Support Systems Lesson Plan, DIESEL-05, Volume 5, Chapter 09, Page 40, Learning Objective 12.a

[3.1/3.2]

064000G010 ..(KA's)

QUESTION: 047 (1.00)

Which of the following describes the purpose of the FIRE EMERGENCY BYPASS switches located on the local control panel?

a. Bypass control room controls.

b. Bypass the SEC load sequencing signals.

c. Bypass control room and cable room wiring to allow local starting of the diesel during a shutdown outside the control room.

d. Bypass Carbon Dioxide fire protection for the diesel generator in order to test fire detectors.

ANSWER: 047 (1.00)

b.

REFERENCE:

Diesel Generator and Support Systems System Description, Volume 4, Chapter 41, page 38.
Diesel Generator and Support Systems Lesson Plan, DIESEL-05, Volume 5, Chapter 09, Learning Objective 14.

[3.4/3.6]

064000G007 ..(KA's)

QUESTION: 048 (1.00)

A RCP has just been started with RCS pressure at 325 psig. Select the reason that a minimum of 200 psid on the RCP seals is required for RCP operation?

a. Ensures that adequate seal cooling flow from the RCS is available.

b. Prevents the #1 RCP seal from swapping from a face rubbing to a film riding seal.

c. Prevents the weight of the seal ring from limiting cooling flow through the seal gap.

d. To avoid having to open the bypass valve around #1 seal.

ANSWER: 048 (1.00)

c.

REFERENCE:

Facility Exam Bank 001334
Reactor Coolant Pump System Description, Volume 1, Chapter 4, page 15.
Reactor Coolant Pump Lesson Plan, RCPUMP-04, Volume 1, Chapter 6,
Learning Objective 13.

[2.8/3.1]

003000K404 ..(KA's)

QUESTION: 049 (1.00)

A reactor cooldown is in progress and the Low Steam Line Pressure SI signal has been manually blocked.

Identify the plant response in the event of a main steam line rupture at the mixing bottle.

a. Both an Safety Injection and Main Steam Line Isolation will occur.

b. A Main Steam Line Isolation will occur.

c. A Main Steam Line Isolation will occur only if pressure decreases to below 600 psig.

d. An Safety Injection will occur but a Main Steam Line Isolation will NOT occur.

ANSWER: 049 (1.00)

b.

REFERENCE:

Reactor Protection System Description, Volume 3, Chapter 28, page 40.
Reactor Protection System Lesson Plan, RXPROT-08, Volume 8, Chapter 2,
Learning Objective 2.

[3.9/4.4]

013000K403 ..(KA's)

QUESTION: 050 (1.00)

Which of the following mode changes for the Subcooling Margin Monitor (SMM) will ONLY occur manually?

a. Shifting from "Normal" to "Adverse" mode on increasing containment pressure.

b. Shifting from "Adverse" to "Normal" mode on decreasing containment pressure.

c. Shifting from "Normal" to "Adverse" mode on increasing containment radiation.

d. Shifting from "Adverse" to "Normal" mode on decreasing containment radiation.

ANSWER: 050 (1.00)

d.

REFERENCE:

Facility Exam Bank 000760
Incore Nuclear Instrumentation System Description, Volume 2, Chapter 21, page 26.
Incore Nuclear Instrumentation System Lesson Plan, INCORE-04, Volume 7, Chapter 02, Learning Objective 5.

[2.9/2.9]

017000G007 ..(KA's)

QUESTION: 051 (1.00)

Core reloading following an outage is in progress. An assembly is being moved from its storage location in the Fuel Handling building to the fuel transfer cart. A loss of power to the Spent Fuel Handling Crane radiation monitor has occurred causing its protective function to occur.

Identify the effect this will have on movement of the crane.

a. All movement of the crane and suspended load is prevented except for downward movement of the suspended load.

b. Upward movement of the suspended load is prevented but all other movement of the crane is available.

c. Upward and downward movement of the suspended load is prevented but the crane can be moved.

d. All movement of the crane and the suspended load is prevented.

ANSWER: 051 (1.00)

a.

REFERENCE:

Radiation Monitoring System, S2.OP-SO.RM-00001, Attachment 1.
Radiation Monitoring System Lesson Plan, RMS000-00, Volume 8, Chapter 12, Learning Objective 5.j.

[3.1/3.5]

072000K302 ..(KA's)

QUESTION: 052 (1.00)

Which of the following would be indication that VCT level detector LT-114 has failed low?

a. VCT Hi-Lo Level alarm is annunciated.

b. Following a down power transient, CV-35 trips open instead of modulating open.

c. Auto Makeup operates continuously.

d. VCT Level indication on the control console indicates downscale.

ANSWER: 052 (1.00)

b.

REFERENCE:

Chemical and Volume Control System Description, Volume 1, Chapter 6, page 22-23.
Chemical and Volume Control System Lesson Plan, CVCS00-03, Volume 1, Chapter 8, Learning Objective 2.e.

[3.2/3.0]

004020A305 ..(KA's)

QUESTION: 053 (1.00)

Following a loss of all off-site power, EDG "2C" fails to start.

Which of the following lists BOTH the control air headers that will NOT have a source of air?

a. 2A and 1A

b. 2A and 2B

c. 1A and 1B

d. 1B and 2B

ANSWER: 053 (1.00)

a.

REFERENCE:

Station and Control Air System Description, Volume 3, Chapter 39, page 13.
Control Air Lesson Plan, 300S-CONAIR-07, Volume 2, Chapter 03, Learning Objective 15.

[3.0/3.4]

078000K303 ..(KA's)

QUESTION: 054 (1.00)

Temperature of the air mixture in the containment hydrogen recombiners is controlled by:

a. restricting the inlet air flow to the recombiner.

b. controlling the air flow at the discharge of the recombiner which is used to preheat the inlet flow.

c. cycling the electric heaters on and off to maintain temperature.

d. adjusting the power to the electric heaters.

ANSWER: 054 (1.00)

d.

REFERENCE:

Containment and Support Systems Lesson Plan, CONTMT-02, Volume 2, Chapter 01, page 111, Learning Objective 2.b

[2.6/3.1]

028000K601 ..(KA's)

QUESTION: 055 (1.00)

A small break loss of coolant accident has occurred inside containment. Select the effect that an increase in containment temperature will have on pressurizer level.

Reference leg density will decrease causing:

a. decreased differential pressure and an increase in level indication.

b. increased differential pressure and an increase in level indication.

c. decreased differential pressure and a decrease in level indication.

d. increased differential pressure and a decrease in level indication.

ANSWER: 055 (1.00)

a.

REFERENCE:

Pressurizer Pressure and Level Control, PZRP/L-00, Volume 7, Chapter 05, page 36, Learning Objective 2.

[3.5/3.6]

011000A101 ..(KA's)

QUESTION: 056 (1.00)

A fuel assembly is being moved from the upender to an incore location. As the assembly is being lowered into the core the spotter recognizes that the assembly is being lowered into the incorrect location.

Select the required action.

a. Stop all movement of the fuel.

b. Move the assembly to its required location.

c. Raise the assembly from the core and then stop fuel movement.

d. Move the assembly to the upender and stop fuel movement.

ANSWER: 056 (1.00)

a.

REFERENCE:

Conduct of Fuel Handling, NC.NA-AP.ZZ-0049, Rev . 0, Page 13.
Discussion with facility management indicated that choice "a" was the expected course of action.
[2.3/2.9]

034000G001 ..(KA's)

QUESTION: 057 (1.00)

During a startup, the High Neutron Flux Reactor Trip and the Control Rod Stop functions on the Intermediate Range Nuclear Instruments are blocked by depressing a pair of pushbuttons on the control board.

How does this affect operation of the channel?

a. The input to the Level trip and Rod Stop bistables are shunted to ground.

b. The output from the Level Trip and Rod Stop bistables is blocked.

c. A bypass circuit around the Level Trip and Rod Stop bistables is shunted closed.

d. The input from the detector to the log current amplifier is blocked.

ANSWER: 057 (1.00)

b.

REFERENCE:

Excore Nuclear Instrumentation Lesson Plan, EXCNIS-07, Volume 7, Chapter 01, Learning Objective 2.f., page 44.

[3.9/4.2]

015000K406 ..(KA's)

QUESTION: 058 (1.00)

Given the following for Unit 2:

The reactor has tripped.
3 control rods have failed to insert on the trip.

Which one of the following is the MINIMUM volume that must be added to the RCS from the RWST as a result of the stuck rods?

a. 1960 gallons

b. 5600 gallons

c. 11200 gallons

d. 16800 gallons

ANSWER: 058 (1.00)

d.

REFERENCE:

3 rods X 80 min/rod X 70 gpm = 16,800 gal.
S2.EOP-TRIP-2, Rev. 3, sheet 1.
Salem LP TRIP02-03, EO 2.a; II.G, page 7.
Facility Bank Mod (001359).
KA [3.5/4.4]

000005A203 ..(KA's)

QUESTION: 059 (1.00)

When raising reactor power, a Control Bank C (CBC) Group 1 rod stops moving at 218 steps and the URGENT FAILURE alarm actuates.

What problem is generated if the operator continues to withdraw rods using individual bank positions (CBC & CBD) on the Bank Selector Switch?

a. The position difference between the affected rod(s) and the remainder of rods in the bank will exceed allowed Technical Specification limits.

b. The RIL Monitor will calculate a higher setpoint for the RIL alarm.

c. A loss of bank overlap in the control groups will occur.

d. The ROD BOTTOM/ROD DROP alarm will NOT clear.

ANSWER: 059 (1.00)

c.

REFERENCE:

S2.OP-AB.ROD-0001(Q) Rev. 2, Technical Bases Document, 2.4 2nd para.
Step 3.5, page 4.
Salem LP ABROD1-01, EO 3; C.1, page 6.
KA [3.6/3.4]

000005A101 ..(KA's)

QUESTION: 060 (1.00)

The plant has been ramping up in power from a shutdown condition and is currently at 90% power. The control board operator notices that ONE rod in CONTROL BANK C is indicating 216 steps with the remainder of the group at 225 steps. The investigating technician reports the problem to be a blown lift coil fuse which continues to fail when replaced due to a bad fuseholder. He reports all other facets of rod control for the rod in question are functioning properly. The technician reports 6-8 hours will be needed to replace the fuse holder.

Which one of the following describes the action required to be taken?

a. Immediately initiate a boration that is equivalent to the worth of the rod at 216 steps.

b. Within one hour, verify the Shutdown Margin (SDM) and be in HOT STANDBY within 6 hours.

c. Within one hour, align the remainder of the rods in the group with the rod at 216 steps.

d. NO administrative action is required since the rod is NOT misaligned greater than 12 steps.

ANSWER: 060 (1.00)

d.

REFERENCE:

Salem Technical Specification Interpretation - 3.1.3.1, Rev. 0.
Salem Tech Specs 3.1.3.1, page 3/4 1-18.
KA [3.1/3.3]

000005G005 ..(KA's)

QUESTION: 061 (1.00)

Given the following for Unit 2:

A small break LOCA has occurred.
The reactor is tripped with safety injection activated.
All ECCS equipment is operating as expected.
All preceding emergency procedure action steps have been performed and currently actions are being directed in 2-EOP-LOCA-2, "POST LOCA COOLDOWN AND DEPRESSURIZATION".

Which one of the following describes actions prescribed in 2-EOP-LOCA-2 which would prevent an Inadequate Core Cooling Condition?

a. Stabilize SG pressure and level, increase ECCS injection and RCS pressure to increase heat removal through the break.

b. Increase heat removal via the SGs to increase cooldown and depressurization of the RCS allowing increased ECCS flow.

c. Reduce ECCS flow and lower RCS pressure to increase heat removal through the break while maintaining a constant SG pressure.

d. Maintain pressurizer level greater than 20% and RCS pressure greater than 1000 psig to ensure subcooling can be maintained while steaming the SGs.

ANSWER: 061 (1.00)

b.

REFERENCE:

Salem LP LOCA02-02, EO 2.b; II.E & G, pages 7-8.
Facility Bank Mod (001381).
KA [4.5/4.7]

000011A210 ..(KA's)

QUESTION: 062 (1.00)

With Unit 2 operating at 100% power, a Large Break LOCA occurs on the 24 Hot Leg. If ALL ECCS supplies were at the Technical Specification MINIMUM values when the accident began, approximately how much cooling water is delivered to the RCS at the time the transition to Cold Leg Recirculation was initiated?

a. 215,000 gallons

b. 239,000 gallons

c. 390,000 gallons

d. 424,000 gallons

ANSWER: 062 (1.00)

b.

REFERENCE:

Selected section of S2.OP-DD.ZZ-OD74(Z) is provided for reference (will be provided to candidates).
RWST (364,500 - 150,000), SI Accumulators (4 x 6223)
= (214,500 + 24,892) = 239,392 gallons
Salem - Unit 2 Technical Specifications 3.5.1.b, page 3/4 5-1; 3.5.5.a, page 3/4 5-9.
2-EOP-LOCA-1 Rev. 4, Continuous Action Summary, sheet 1 B3.
OD-74 (S2.OP-DD.ZZ-OD74(Z)) Rev.2 various locations.
Salem LP LOCA01-02, EO 4.
KA [4.1/4.2]

000011A113 ..(KA's)

QUESTION: 064 (1.00)

The reactor is at 100% power. All charging flow is lost and the crew is unable to restore a charging pump. During the attempts to restore charging, an operator inadvertently misunderstands an order and closes valve CC131. The valve fails to reopen and the SRO enters AB.CC-0001 "COMPONENT COOLING WATER ABNORMALITY".

Which of the following statements describes the general actions required by this procedure?

a. If RCP motor and pump bearing temperatures exceed 175 degrees F, reduce power to less than 36% and stop all RCPs.

b. If seal flow cannot be restored within 5 minutes, trip the reactor and stop all RCPs.

c. Immediately trip the reactor and stop all RCPs.

d. Reduce power to less than 36% and stop all RCPs.

ANSWER: 064 (1.00)

c.

REFERENCE:

S2.OP-AB.CC-0001 Rev.01, Step 3.1
Salem LP ABCC01-01, EO 2; III.C.1, page 5.
Facility Bank Mod(001474).
KA [3.3/3.3]

000026G012 ..(KA's)

QUESTION: 065 (1.00)

Unit 2 was operating at 100% power. A reactor trip was required but did NOT occur. SI is NOT required. The transition to 2-EOP-FRSM-1 "RESPONSE TO NUCLEAR POWER GENERATION" was made.

What is the PREFERRED method for adding negative reactivity to the core assuming a reactor trip is unsuccessful?

a. Rapid boration through CVCS from the BATs via Boric Acid Pumps.

b. Injection via the BIT from the RWST through CVCS.

c. Manual actuation of SI.

d. Heatup of the primary coolant resulting from the turbine trip.

ANSWER: 065 (1.00)

a.

REFERENCE:

2-EOP-FRSM-1 Rev. 3, Steps 1-3, sheet 1 C-L4 & B-H9.
2-EOP FRSM-1 ERG BASIS Rev. 3, Steps 1-4 Basis (2nd para.), page 8.
Salem LP FRSM01-01, EO 6.a; III.A - C, page 9.
KA [4.5/4.5]

000029G010 ..(KA's)

QUESTION: 066 (1.00)

2-EOP-FRSM-1 "RESPONSE TO NUCLEAR POWER GENERATION" directs the operator to reduce RCS pressure using pressurizer PORV's if pressure is above 2335 psig. Which one of the following describes the purpose for reducing RCS pressure.

a. To allow increased boration rate from the charging system.

b. To prevent a high pressure inadequate core cooling situation.

c. To prevent a small break LOCA situation from developing from a failed open pressurizer code safety valve.

d. To allow maximum negative reactivity addition from voiding.

ANSWER: 066 (1.00)

a.

REFERENCE:

S2.EOP-FRSM-1 Rev.03
Salem LP FRSM01-01, EO 2; II.D.1, page 9.
Facility Bank Mod(000896).
KA [4.4/4.7]

000029K312 ..(KA's)

QUESTION: 067 (1.00)

In the EOP network, if RCS pressure is less than 1355 psig AND either a CCP or SI pump is running and providing at least 100 GPM, then all RCPs are stopped.

Which one of the following describes the BASIS for this step?

a. To protect the RCP seals.

b. To minimize mass loss from a small break LOCA.

c. To assist establishment of natural circulation.

d. To minimize RCS heatup from energy added by the RCPs.

ANSWER: 067 (1.00)

b.

REFERENCE:

2-EOP-LOSC-1RCP, Rev. 4.
Salem LP LOSC02-01, EO 2; II.C & F.2, pages 8-9.
Salem LP RCPTAA-01, EO 2; II.B.2.i.2) & j, pages 9-10 & III.A.2, page 11.
Facility Bank Mod (000390).
KA [4.5/4.7]

000040K304 ..(KA's)

QUESTION: 068 (1.00)

The reactor is at 100% power and excessive steam flow has been diagnosed. Which of the following condition or conditions require initiation of a steam line isolation and entry into the EOP network (reactor trip) while performing S2.OP-AB.STM-0001(Q) "EXCESSIVE STEAM FLOW"?

1. Reactor power increasing uncontrollably.
2. RCS cooldown rate of 50 degrees F per hour.
3. Control Rods begin to step 'OUT' in fast speed.
4. The leak is determined to be inside containment.

a. 1 only

b. 1 and 2 only

c. 1, 2, and 3 only

d. 1, 2, 3, and 4.

ANSWER: 068 (1.00)

b.

REFERENCE:

S2.OP-AB.STM-0001 Rev. 2; step 3.1, page 1.
Salem LP ABSTM1-01, EO 2; III.C.1, page 6.
Facility Bank Mod (001030).
KA [4.4/4.7]

000040A204 ..(KA's)

QUESTION: 069 (1.00)

Unit 2 is operating at 90% power when condenser vacuum begins to decrease unexpectedly. A power reduction is directed in accordance with S2.OP-AB.COND-0001(Q) "LOSS OF CONDENSER VACUUM". The procedure directions limit the turbine load reduction ramp rate to 5% per minute or less.

Why should load rates not exceed 5%/min?

a. Prevents operation of the Steam Dumps.

b. Allows starting of ALL Condenser Vacuum Pumps.

c. Minimizes wear on the low pressure turbine last row blading.

d. Ensures Condensate Pump suction temperatures remain less than that required for NPSH.

ANSWER: 069 (1.00)

a.

REFERENCE:

S2.OP-AB.COND-00001(Q) Rev. 1; 3.0 NOTE, page 1.
S2.OP-AB.COND-00001(Q) TBD Rev. 1; 2.4, page 3.
Salem LP ABCOND-01, EO 3; III.C.1, page 6.
KA [3.9/4.1]

000051A202 ..(KA's)

QUESTION: 070 (1.00)

Transition has been made from 2-EOP-LOPA-1 "LOSS OF ALL AC POWER" to 2-EOP-LOPA-3 "LOSS OF ALL AC POWER RECOVERY/SI REQUIRED" when the NSS is notified that a RED PATH exists for Heat Sink CFST and a PURPLE PATH exists for Core Cooling CFST.

Which action should the NSS take for this condition?

a. Transition to 2-EOP-TRIP-1 "REACTOR TRIP OR SAFETY INJECTION" and perform actions beginning with Step 1.

b. Transition to 2-EOP-FRCC-2 "RESPONSE TO DEGRADED CORE COOLING" and direct actions establish SI flow.

c. Transition to 2-EOP-FRHS-1 "RESPONSE TO LOSS OF SECONDARY HEAT SINK" and direct actions to restore feed to the SGs.

d. Continue to direct actions of 2-EOP-LOPA-3 "LOSS OF ALL AC POWER RECOVERY/SI REQUIRED" for Safeguards Systems status.

ANSWER: 070 (1.00)

d.

REFERENCE:

2-EOP-LOPA-3 Rev. 2; sheet 1, CAUTION 1.
Salem LP LOPA00-02, EO 9; I.C.1, page 8 & IV.A, page 11 & VIII.C, page 19.
KA [4.1/4.1]

000055G011 ..(KA's)

QUESTION: 071 (1.00)

Which one of the following describes equipment that is sequenced differently by the Safeguards Equipment Control (SEC) System for a LOPA if an SI signal is present compared to a situation without an SI signal present (LOPA only)?

a. 22 Auxiliary Feed Pump and 22 Chiller

b. 22 Component Cooling Pump and 22 RHR Pump

c. 24 SW Pump and 22 Auxiliary Feed Pump

d. 22 RHR Pump and 22 Chiller

ANSWER: 071 (1.00)

b.

REFERENCE:

2-EOP-LOPA-1 Rev. 2; sheet 1, Table A and Table B.
Salem LP SEC000-01, EO 4; , IV.A.1.c, page 24 and IV.B1.e, page 27 and IV.c, pages 27-28.
KA [4.1/4.5]

000055A106 ..(KA's)

QUESTION: 072 (1.00)

The following Unit 2 conditions exist:

- 50% power
- Two main feed pumps in service
- All systems in automatic
- 115 VAC vital bus 2A has just de-energized

Which ONE of the following describes the IMMEDIATE ACTIONS required by S2.OP.AB-115-0001(Q) "LOSS OF 2A 115V VITAL INSTRUMENT BUS"?

a. Take manual control of ALL Feedwater Valves (BF19s), and place the Rod Bank Selector Switch in MANUAL.

b. Take manual control of ALL Feedwater Valves (BF19s), and depress the Train 'B' "OFF & RESET BYPASS TAVG" pushbutton for Steam Dump control.

c. Trip the reactor, and start both motor-driven Auxiliary Feedwater pumps.

d. Trip the reactor, and depress the Train 'B' "OFF & RESET BYPASS TAVG" pushbutton for Steam Dump control.

ANSWER: 072 (1.00)

a.

REFERENCE:

S2.OP-AB.115-0001(Q) Rev. 5, 2.1 & 2.2, page 1.
Salem LP AB115V-01, EO 1; III.B.1, page 6.
Facility Bank Mod (001275).
KA [3.5/3.5]

000057A106 ..(KA's)

QUESTION: 073 (1.00)

In S2.OP-AB.115-0002, "Loss of 2B 115V Vital Instrument Bus", prior to Step 3.2, a caution states,

"A Reactor Trip will occur if any Steam Generator narrow range level decreases to equal to or less than 25%"

The reason for this trip is that the loss of power to the vital instrument bus:

a. did nothing to change the protection logic since a single SG level reaching less than 25% is the normal reactor trip logic.

b. caused a channel II low level signal which if combined with any second channel in that SG reaching less than 25% will cause a reactor trip.

c. caused a channel II low pressure signal which if combined with a channel III or IV low level of 25% on any SG will cause a reactor trip.

d. caused a channel II steam flow/feed flow mismatch which if combined with a channel III or IV low level of 25% on any SG will cause a reactor trip.

ANSWER: 073 (1.00)

d.

REFERENCE:

S2.OP-AB.115-0002(Q) Rev. 5; CAUTION 3.2, page 1.
Salem LP AB1152-01, EO 3; III.C.3, page 7.
RPS Logic Diagram 221056 B 9545-7; sheet 7, LOW FEEDWATER FLOW.
Facility Bank Mod (001494).
KA [3.7/3.9]

000057A203 ..(KA's)

QUESTION: 074 (1.00)

A primary to secondary leak has developed in the 21 SG. The Steam Generator Blowdown monitor 2R19A indicates a steady but slow rise in detected radiation.

What automatic action(s) occur when the WARNING function occurs?

a. All SG Blowdown Outlet Valves (21-24 GB4) close and the 3-Way Discharge Valves (2GB74 and 2GB112) align to the Waste Monitor Holdup Tank.

b. Outlet valve from 21 SGBT (2GB50), Blowdown Flow Control Valves (21-24 GB10) and Blowdown Discharge Valves to 22 Condenser (21-24 GB185) close.

c. The SG Blowdown Outlet Valve (21GB4), Blowdown Flow Control Valve (21GB10) and Blowdown Discharge Valve to 22 Condenser (21GB185) close.

d. The Blowdown Disch Valve to 22 Condenser (21GB185) closes and the 3-Way Discharge Valves (2GB74 and 2GB112) align to the Waste Monitor Holdup Tank.

ANSWER: 074 (1.00)

b.

REFERENCE:

S2.OP-AB.RAD-0001(Q) Rev. 5, ATTACHMENT 1, 1.E.1, page 7.
Salem LP ABRAD1-00, EO 2; II.C.3, page 7.
Salem LP RMS000-00, EO 2.e & 5.g; Table 2, page 83.
KA [3.6/3.9]

000059A205 ..(KA's)

QUESTION: 075 (1.00)

Given the following:

Loss of offsite power has occurred.
Diesel Generator 1A fire protection system has actuated.
Alarms in the control room indicate there is a fire in the 1B Diesel Generator room.
The "First In With Lockout" signal is preventing CO2 deluge actuation in the 1B Diesel Room.

What must occur in order to actuate the 1B Diesel Generator CO2 deluge system?

a. The deluge system for the 1A Diesel must be manually isolated.

b. Manually operate the pilot cabinet operating lever for the 1B Diesel actuation system to bypass the electrical circuit.

c. Allow the timed discharge sequence circuitry on the Fire Protection system to cycle until it picks up a new signal.

d. Operate the toggle switch at the fire protection panel to energize the shunt trip coil for the 1B Diesel actuation system.

ANSWER: 075 (1.00)

b.

REFERENCE:

Salem LP FIRPRO-03, EO 7.b; IV.G.9, pages 71-72.
Facility Bank Mod (001348).
KA [3.4/3.7]

000067A108 ..(KA's)

QUESTION: 076 (1.00)

What action is performed when the Control Room is evacuated in accordance with S2.OP-AB.CR-0001(Q) "CONTROL ROOM EVACUATION" but is NOT performed when it is evacuated in accordance with S2.OP-AB.CR-0002(Q) "CONTROL ROOM EVACUATION DUE TO FIRE IN CONTROL ROOM, RELAY ROOM, OR CEILING OF 460/230V SWITCHGEAR ROOM"?

a. The 21 and 23 Reactor Coolant Pumps are tripped.

b. The 21 and 22 Auxiliary Feedwater Pumps are started.

c. Lockout Switches on 2RP4 are placed in OPERABLE position.

d. Pressurizer PORVs 2PR6 and 2PR7 are placed in CLOSED position.

ANSWER: 076 (1.00)

c.

REFERENCE:

S2.OP-AB.CR-0001(Q) Rev. 2, Step 3.1 item E, pages 1-2.
S2.OP-AB.CR-0002(Q) Rev. 3, Step 3.1, pages 1-2.
Salem LP ABCR01-01, EO 3.A & C; III.C.4, page 8 and IV.C.5, pages 10-11.
KA [3.9/4.1]

000068A121 ..(KA's)

QUESTION: 077 (1.00)

Which one of the following controls is located on the Hot Shutdown Panel 213?

a. Aux. Feedwater Storage Tank Makeup Valve OPEN/CLOSE switch.

b. Main Steam Isolation Valves OPEN/CLOSE switches.

c. Pressurizer Heater Backup Group ON/OFF switches.

d. Centrifugal Charging Pumps START/STOP switches.

ANSWER: 077 (1.00)

d.

REFERENCE:

S2.OP-AB.CR-0001(Q) Rev. 2, Step 3.6, page 2 (MSIV); Step 3.7, page 4 (Prz Heaters, AFST MU); and 3.9.A.1, page 4.
Salem LP ABCR01-01, EO 4.A; I.B, page 7.
KA [3.9/4.0]

000068K201 ..(KA's)

QUESTION: 078 (1.00)

In S2.OP-AB.CR-0002 "CONTROL ROOM EVACUATION DUE TO FIRE IN CONTROL ROOM, RELAY ROOM, OR CEILING OF 460/230V SWITCHGEAR ROOM", many actions are performed prior to leaving the Control Room. Even though the tasks may have been completed, the procedure later requires the tasks be completed AGAIN remotely, with the exception of tripping the reactor.

The MAIN reason these tasks are repeated is:

a. only ONE train of systems and components used to achieve and maintain Hot Shutdown conditions must be free of fire damage.

b. non-Safety related associated circuits can sustain fire damage that can affect exposed safe shutdown circuits.

c. the only Operator action taken credit for in the analysis before Control Room evacuation is Reactor Trip.

d. complete isolation from the control room is mandatory for ALL Safe Shutdown equipment circuitry.

ANSWER: 078 (1.00)

c.

REFERENCE:

S2.OP-AB.CR-0002(Q) Rev. 3, Technical Bases Document, 2.1, page 5.
Salem LP ABCR01-01, EO 3.B; IV.C.6, page 11.
Facility Bank Mod (001501).
KA [4.1/4.5]

000068K312 ..(KA's)

QUESTION: 079 (1.00)

Which ONE of the following statements describes the reason for stopping ALL RCPs at step 6 of EOP-FRCE-1 "RESPONSE TO EXCESSIVE CONTAINMENT PRESSURE"?

a. Prevent uncovery of the core due to separation of the steam-water mixture if the RCPs were to subsequently trip.

b. Prevent exceeding containment design pressure due to forced pumping of the RCS and ECCS inventory out the break.

c. Remove the RCP heat load from containment since the containment spray system is designed for conditions with the RCPs stopped.

d. Remove the RCPs from service for pump protection since CCW cooling flow is isolated to and from the RCPs.

ANSWER: 079 (1.00)

d.

REFERENCE:

2-EOP FRCE-1 ERG BASIS Rev. 2; Step 3-5 Basis, page 8.
Salem LP FRCE00-02, EO 3; II.C.6, page 10.
Facility Bank Mod (000186).
KA [3.8/4.2]

000069K301 ..(KA's)

QUESTION: 080 (1.00)

During response to inadequate core cooling, the Control Room operators have been unsuccessful in reducing core exit thermocouples to LESS THAN 1200 degrees F. EOP-FRCC-1 "RESPONSE TO INADEQUATE CORE COOLING" directs the operators to start RCP(s) in an attempt to reduce core exit temperature.

If RCP startup conditions CANNOT be established, which one of the following statements is applicable?

a. ALL RCPs will be started sequentially if core exit thermocouple temperatures remain above 1200 degrees F.

b. Only ONE RCP may be started, then normal startup conditions must be established prior to starting additional RCPs.

c. RCPs should be started only after permission is obtained from the Emergency Duty Officer.

d. RCPs will NOT be started without normal startup conditions because severe damage to the RCP could result in higher core temperatures.

ANSWER: 080 (1.00)

a.

REFERENCE:

2-EOP-FRCC-1 Rev. 3; Steps 29-31.1, sheet 3 L10 & A-F17.
2-EOP FRCC-1 ERG BASIS Rev. 5; FRCC-1 Step 29 & 30-31, pages 53-55.
Salem LP FRCC00-02, EO 2.2; III.M & N & O, page 10.
KA [3.6/3.8]

000074K201 ..(KA's)

QUESTION: 081 (1.00)

An inadequate core cooling condition exists for Unit 2 with core exit thermocouples at 1400 degrees F and RVLIS indication at 33% and decreasing. A RCP is started in an attempt to establish temporary cooling to the core by:

a. natural single phase convection.

b. forced single phase radiation.

c. forced two phase flow.

d. natural two phase conduction.

ANSWER: 081 (1.00)

c.

REFERENCE:

2-EOP FRCC-1 ERG BASIS Rev. 5; FRCC-1 Step 30-31, pages 54.
Salem LP FRCC00-02, EO 3; III.C, page 8-9.
KA [4.5/4.9]

000074K103 ..(KA's)

QUESTION: 082 (1.00)

Unit 2 reactor trips from 100% power with SI actuation due to a steam line break downstream of the Main Steam Isolation Valves (21-24 MS167). The control room operators are taking actions as directed in the EOP network and have obtained the required/expected plant response, except for the following:

21MS167 will not close
Total attainable AFW flow is 20E4 lbm/hr
Current SG parameters are as follows:
#21 - 8% narrow range; 500 psig
#22 - 15% narrow range; 900 psig
#23 - 12% narrow range; 925 psig
#24 - 9% narrow range; 975 psig

What is the impact of this configuration on secondary heat removal capability?

a. Secondary heat removal capability is adequate as long as current SG levels are maintained.

b. The loss of secondary heat removal capability is imminent, unless SG #21 is isolated.

c. Secondary heat removal capability is lost because AFW flow is inadequate.

d. Secondary heat removal capability is lost because SG levels are inadequate.

ANSWER: 082 (1.00)

a.

REFERENCE:

2-EOP-CFST-1 (FRHS-1,2,3,4,5), Entry Conditions to FRHS series.
Salem LP FRHS01-02, EO 1; II.A, page 10; IV.A.1, page 15; VI.A, page 16; VIII.A.1, page 17; X.A.1, page 18.
Facility Bank Mod (000081).
KA [3.7/4.1]

000007K106 ..(KA's)

QUESTION: 083 (1.00)

Unit 2 was at 50%. S2.OP-AB.RCP-0001(Q) "REACTOR COOLANT PUMP ABNORMALITY" was entered and a power decrease has initiated with all systems in automatic due to 24 RCP problems.

Current plant conditions:

Reactor Power - 34%
Pzr level - 45%
Pzr pressure - 2245 psig
SG levels at program

Assuming NO operator action, the 24 RCP trips and a reactor trip follows. What is the cause of the reactor trip?

a. Low RCS flow in ONE loop since reactor power was above the P-8 setpoint.

b. Pressurizer pressure reached the trip setpoint since spray flow was lost.

c. Shrink in the 24 SG caused level to decrease below the trip setpoint.

d. RCS heatup caused Pressurizer level to reach the trip setpoint.

ANSWER: 083 (1.00)

c.

REFERENCE:

S2.OP-AB.RCP-0001(Q) Rev. 2, ATTACHMENT 1 B.1, page 9.
Salem LP ABRCP1-01, EO 4.A; I.B, page 5.
KA [3.6/3.7]

000007A104 ..(KA's)

QUESTION: 084 (1.00)

As a MINIMUM, what is required to be checked to confirm a reactor trip upon entry into 2-EOP-TRIP-1 "REACTOR TRIP OR SAFETY INJECTION"?

a. IRPI indicate rods on bottom, and REACTOR TRIP light on RP4 lit.

b. IRPI indicate rods on bottom, and Power Range neutron flux is less than 5%.

c. Reactor trip breaker and associated bypass breaker open, and REACTOR TRIP LITE on RP4 lit.

d. Reactor trip breaker and associated bypass breaker open, and Power Range neutron flux is less than 5%.

ANSWER: 084 (1.00)

a.

REFERENCE:

Salem LP TRIP01-05, EO 3.c; III, page 7.
KA [4.3/4.5]

000007A206 ..(KA's)

QUESTION: 085 (1.00)

Which one of the following indicates that the PRT rupture disk is ruptured following a pressurizer PORV failing OPEN and NOT isolated?

a. Pressurizer level decreases.

b. PRT temperature decreases.

c. Actual PRT level decreases.

d. Relief line temperature increases.

ANSWER: 085 (1.00)

b.

REFERENCE:

Salem LP LOCATA-00, EO 3; E.3.c, page 15.
KA [3.2/3.7]

000008K101 ..(KA's)

QUESTION: 086 (1.00)

During the initial stage of a Small Break LOCA, Pressurizer level is a measure of the liquid mass in the RCS except if the break is in the vapor space of the Pressurizer vessel.

If the leak is in the Pressurizer vapor space,:

a. the level instrumentation is NOT environmentally qualified for the conditions in the Pressurizer compartment.

b. rapid evaporation causes superheat conditions in the vapor space above the liquid surface.

c. injection of much cooler ECCS water results in a large instrumentation error.

d. saturation conditions creates voids in the reactor vessel head and hot legs.

ANSWER: 086 (1.00)

d.

REFERENCE:

Transient and Accident Analysis Book, Chapter 8 SBLOCA; MODE 1 Decay Heat by Natural Circulation with the Pressurizer Controlling Pressure, page 2-11.
Salem LP LOCATA, EO 3.
KA [3.7/4.4]

000008K301 ..(KA's)

QUESTION: 087 (1.00)

In response to an RCS leak, S2.OP-AB.RC-0001(Q) "REACTOR COOLANT SYSTEM LEAK" directs the operator to transfer to ONE centrifugal charging pump in service and to reduce letdown flow to ONE 45 gpm orifice. If Pressurizer level cannot be maintained in this arrangement, then SI should be initiated.

Salem management has determined that SI actuation is required with these conditions because the leak rate at normal operating pressure is greater than approximately:

a. 45 gpm.

b. 75 gpm.

c. 100 gpm.

d. 150 gpm.

ANSWER: 087 (1.00)

c.

REFERENCE:

S2.AB.RC-0001(Q) Rev. 0 Technical Bases Document, 2.4 1st para., page 4.
Facility Bank Mod (0001036).
KA [4.2/4.5]

000009K321 ..(KA's)

QUESTION: 088 (1.00)

Following an SI actuation due to a Small Break LOCA condition, RCS pressure stabilizes at 1050 psig. Which set of parameters would be monitored to determine if an inadequate core cooling condition existed?

a. SG pressures and Core delta-T

b. RVLIS Dynamic Range and Subcooling Margin

c. RCS pressure and RCS wide range Thot temperatures

d. Core Exit Thermocouple temperatures and RVLIS Full Range

ANSWER: 088 (1.00)

d.

REFERENCE:

2-EOP-Trip-1 Rev. 4; Continuous Action Summary, sheet 2 D3.
Salem LP TRIP1-05, EO 3 & 6; VIII, page 13 & TABLE 1.
Facility Bank Mod (000113).
KA [4.3/4.7]

000009A239 ..(KA's)

QUESTION: 089 (1.00)

Unit 2 is at 100% with the following conditions:

Pressurizer level - 60% decreasing
VCT level - 55% increasing
A11 SEAL WATER FLOW LO bezel alarms lit.
LTDWN HX OUT TEMP HI alarm (OHA E-41) lit.

Which one of the following explains these conditions?

a. Pressurizer PORV open.

b. Letdown isolation.

c. Small break LOCA.

d. Loss of charging flow.

ANSWER: 089 (1.00)

d.

REFERENCE:

SP-206 Appendix 11, Malfunction # MTP-034, 23 CHARGING PUMP TRIP, E. General Sequence, page 3.
S2.OP-AR.ZZ-0011(Q) Rev. 0; Bezel 2-5 through 2-8 SEAL WATER FLOW LO.
S2.OP-AR.Z-0005(Q) Rev. 1: Window E-41 LTDWN HX OUT TEMP HI.
KA [3.2/3.7]

000022A202 ..(KA's)

QUESTION: 090 (1.00)

An accident has occurred on Unit 2 with voids identified in the reactor vessel head. The operators are responding to the condition via EOP-FRCI-3. They have progressed to the point where conditions are being established to start a RCP. The procedure requires an initial Pressurizer level of at least 59% prior to the pump start.

What is the reason for achieving this much RCS inventory prior to the RCP start?

This Pressurizer (Prz) level will:

a. ensure sufficient NPSH is provided for the RCP with potential void conditions in the loops.

b. facilitate collapsing voids in the upper head by pressurizing the RCS.

c. ensure natural circulation capability is maintained if the RCP fails to start.

d. accommodate Prz level decrease when voids collapse once the RCP is started.

ANSWER: 090 (1.00)

d.

REFERENCE:

2-EOP-FRCI-3 ERG BASIS Rev. 4; Step 13/16 BASIS 2nd para., page 20.
Salem LP FRCI00-01, EO 2 & 3; IV.D.12, page 19-20.
Facility Bank Mod (000137).
KA [3.5/3.8]

000022K302 ..(KA's)

QUESTION: 091 (1.00)

Given the following Unit 2 conditions:

Mode 5 with the 21 RHR pump in service for cooldown		
Time after reactor shutdown	-	100 hours
RCS pressure	-	300 psig
RCS Tavg	-	150 degrees F
Pressurizer level	-	10% (cold calibrate) stable
SG levels	-	70% stable for each

If the 21 RHR pump was stopped due to indications that the pump became gas bound, what is the preferred method to restore cooling to the RCS in accordance with S2.OP-AB.RHR-0001(Q) "LOSS OF RHR"?

a. Align and start the 22 RHR Pump at full flow.

b. Use reflux cooling with ALL Steam Generators.

c. Initiate Cold Leg Injection using BOTH SI Pumps.

d. Initiate Hot Leg Injection using ONE Centrifugal Charging Pump.

ANSWER: 091 (1.00)

a.

REFERENCE:

S2.OP-AB.RHR-00001(Q) Rev. 2, Technical Basis Document, 2.4 4th para., page 7.
Salem LP ABRHRO-01, EO 2; III.C, page 6.
KA [3.1/3.4]

000025K301 ..(KA's)

QUESTION: 092 (1.00)

Unit 2 is in MODE 4 with RHR system providing the shutdown cooling. All systems are in normal lineup for these conditions.

Valve 2RH25 RHR discharge relief starts to discharge (lift). Which of the following indications would you expect to receive as a result of this relief valve lifting, assuming NO operation action is taken?

a. Aux Bldg sump level increasing.

b. Console alarm "PRT LEVEL HI-LO".

c. Console alarm "VCT PRESSURE HI-LO".

d. RCDT pump runs showing on the alarm typewriter.

ANSWER: 092 (1.00)

b.

REFERENCE:

Salem LP PZRPRT-06, EO 9.e; II.G.5.e, page 18.
Facility Bank Mod (000407).
KA [3.2/3.2]

000025K202 ..(KA's)

QUESTION: 093 (1.00)

Unit 1 reactor is critical at the point of adding heat with bank D at 108 steps. Source Range Nuclear Instrument channel N31 has just failed. Which statement describes the action that must be taken?

a. The failed channel must be repaired before the startup may continue.

b. A Rapid Boration must be initiated with charging flow of at least 33 gpm.

c. Shutdown Margin must be verified to be within allowable limits within 4 hours.

d. NO action is required, the startup may continue.

ANSWER: 093 (1.00)

d.

REFERENCE:

Salem - Unit 1 Technical Specification 3.3.1.1 TABLE 3.3-1 FU 6.A ACTION 4.b, pages 3/4 3-2 & 3/4 3-6.
Salem LP AB.NIS-0001, EO 4.A; III.C.2, page 5.
KA [2.6/3.3]

000032G003 ..(KA's)

QUESTION: 094 (1.00)

Which one of the following describes a condition for a SG Tube Leak that requires initiation of Safety Injection once the plant shutdown has been started?

a. Level difference between the affected SG and the other SGs exceeds 5%.

b. Steam flow/Feed flow mismatch exceeds 1% indicated difference for any SG.

c. 2R15 Condenser Air Ejector Monitor and any 2R19 SG Blowdown Monitor in alarm.

d. Charging Pump suction aligned to the RWST is required due to inadequate VCT level.

ANSWER: 094 (1.00)

d.

REFERENCE:

S2.OP-AB.SG-0001(Q) Rev. 1 Technical Bases Document, 2.4 4th para. (Step 3.2), page 6.
Salem LP ABSG01-01, EO 3; III.C.1, page 7.
KA [3.3/4.1]

000037A212 ..(KA's)

QUESTION: 095 (1.00)

EOP-SGTR-1 is being performed in response to a tube rupture on #23 SG. The cooldown has just been completed but the target temperature value selected by the operators was higher than that stipulated in the procedure.

This error could result in which one of the following conditions?

a. Loss of RCS subcooling before RCS and ruptured SG pressures are equalized.

b. Increase in pressure of the ruptured SG with resultant lifting of the 23 MS Safety Valve.

c. Decrease the time for termination of the primary to secondary leakage.

d. Filling the Pressurizer solid during the subsequent depressurization.

ANSWER: 095 (1.00)

a

REFERENCE:

2-EOP SGTR-1 ERG BASIS Rev. 6, SGTR 1 Step 18, page 46 & 48.
Salem LP SGTR01-04, EO 2.h & 5; III.A.2, page 9 & III.C.14, page 12.
KA [4.0/4.3]

000038K305 ..(KA's)

QUESTION: 096 (1.00)

Unit 2 is at 45% power with the 21 Feedwater Pump in service and the 22 Feedwater Pump warmed up and running at idle. What IMMEDIATE ACTION is required if the 21 Feedwater Pump trips?

a. Trip the reactor.

b. Reduce turbine load to less than 40%.

c. Open the Bypass Polisher Valves 21-23 CN108.

d. Use the SGFP Master in MANUAL to raise the 22 SGFP speed.

ANSWER: 096 (1.00)

c.

REFERENCE:

S2-OP-AB.CN0001(Q) Rev. 5; 2.2, page 1.
Salem LP ABCN01-01, EO 1; III.B.1.b, page 6.
KA [3.2/3.2]

000054G010 ..(KA's)

QUESTION: 097 (1.00)

A loss of Control Air is in progress on Unit 2. The operator is directed by S2.OP-AB.CA-0001(Q) "LOSS OF CONTROL AIR" to start the #2 Emergency Air Compressor (EAC) if 2B header pressure falls below a specific value. This pressure, which also coincides with the automatic start setpoint for the EAC, is at approximately:

a. 105 psig.

b. 90 psig.

c. 85 psig.

d. 70 psig.

ANSWER: 097 (1.00)

c.

REFERENCE:

S2.OP-AB.CA-0001(Q) Rev. 2, Step 3.5-3.7, pages 1 & 2.
Salem LP ABCA01-01, EO 4.B; II.B, page 5.
Salem LP CONAIR-07, EO 6; IV.A.1.c 2nd para., page 21.
KA [3.5/3.4]

000065A104 ..(KA's)

QUESTION: 098 (1.00)

A fuel assembly has been damaged during lifting of the assembly from the Upender in the Fuel Handling Building. The assembly has been fully withdrawn from the Upender frame. Where is the damaged assembly to be placed for protective measures?

a. The bottom of the Transfer Canal.

b. In the Upender which is then lowered.

c. In the Upender, lowered and fully traversed to Containment.

d. In the nearest available location in the Spent Fuel Pool.

ANSWER: 098 (1.00)

d.

REFERENCE:

S2.OP-AB.FUEL-0001(Q) Rev. 0, Step 3.2 & 3.3, page 1.
S2.OP-AB.FUEL-0001(Q) Rev. 0 Technical Basis Document, 2.3, page 4.
Salem LP AFUEL1-02, EO 2 & 4.A; II.A, page 5.
KA [3.2/3.5]

000036G007 ..(KA's)

A N S W E R K E Y

MULTIPLE CHOICE

001	c
002	c
003	d
004	d
005	a
006	c
007	c
008	b
009	d/c
010	d
011	c
012	d
013	d
014	c
015	b
016	b
017	c
018	d
019	a
020	b
021	d
022	b
023	d
024	b
025	b
026	a
027	b
028	a
029	c
030	d
031	c
032	b
033	d
034	a
035	a
036	b
037	c
038	c
039	c
040	b
041	b
042	d
043	c
044	d
045	a

A N S W E R K E Y

046 b

MULTIPLE CHOICE

047 b
048 c
049 b
050 d
051 a
052 b
053 a
054 d
055 a
056 a
057 b
058 d
059 c
060 d
061 b
062 b
063 b
064 c
065 a
066 a
067 b
068 b
069 a
070 d
071 b
072 a
073 d
074 b
075 b
076 c
077 d
078 c
079 d
080 a
081 c
082 a
083 c
084 a
085 b
086 d
087 c
088 d
089 d
090 d

A N S W E R K E Y

091 a

MULTIPLE CHOICE

092 b

093 d

094 d

095 a

096 c

097 c

098 d

099 c

100 c

(********** END OF EXAMINATION **********)

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
001	1.00	9000003
002	1.00	9000005
003	1.00	9000006
004	1.00	9000007
005	1.00	9000009
006	1.00	9000010
007	1.00	9000011
008	1.00	9000012
009	1.00	9000013
010	1.00	9000014
011	1.00	9000015
012	1.00	9000017
013	1.00	9000019
014	1.00	9000020
015	1.00	9000021
016	1.00	9000022
017	1.00	9000023
018	1.00	9000025
019	1.00	9000027
020	1.00	35915
021	1.00	9000029
022	1.00	9000031
023	1.00	9000032
024	1.00	36160
025	1.00	9000036
026	1.00	9000037
027	1.00	9000038
028	1.00	9000039
029	1.00	9000040
030	1.00	9000041
031	1.00	9000043
032	1.00	9000044
033	1.00	9000045
034	1.00	9000047
035	1.00	9000048
036	1.00	9000049
037	1.00	9000050
038	1.00	9000051
039	1.00	9000054
040	1.00	9000055
041	1.00	9000058
042	1.00	9000059
043	1.00	35916
044	1.00	9000061
045	1.00	9000062
046	1.00	9000063
047	1.00	9000064
048	1.00	9000065
049	1.00	35911

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
050	1.00	9000067
051	1.00	9000069
052	1.00	9000070
053	1.00	9000073
054	1.00	9000074
055	1.00	9000075
056	1.00	9000077
057	1.00	9000078
058	1.00	9000080
059	1.00	9000081
060	1.00	9000082
061	1.00	35392
062	1.00	9000084
063	1.00	9000085
064	1.00	9000086
065	1.00	9000088
066	1.00	9000089
067	1.00	9000090
068	1.00	9000091
069	1.00	9000092
070	1.00	9000094
071	1.00	9000095
072	1.00	9000096
073	1.00	9000097
074	1.00	9000098
075	1.00	9000099
076	1.00	9000100
077	1.00	9000101
078	1.00	9000102
079	1.00	9000103
080	1.00	9000104
081	1.00	9000105
082	1.00	9000106
083	1.00	9000107
084	1.00	9000108
085	1.00	35374
086	1.00	9000110
087	1.00	9000111
088	1.00	9000112
089	1.00	31479
090	1.00	9000114
091	1.00	9000115
092	1.00	9000116
093	1.00	9000117
094	1.00	9000118
095	1.00	9000120
096	1.00	9000121
097	1.00	9000122
098	1.00	9000123

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y Q u e s t i o n N u m b e r

QUESTION	VALUE	REFERENCE
099	1.00	9000125
100	1.00	9000126
	100.00	
	100.00	

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

PLANT WIDE GENERICS

QUESTION	VALUE	KA
008	1.00	194001A103
016	1.00	194001A103
011	1.00	194001A103
012	1.00	194001A109
010	1.00	194001A111
009	1.00	194001A112
004	1.00	194001A112
014	1.00	194001A116
013	1.00	194001A116
015	1.00	194001A116
002	1.00	194001K101
003	1.00	194001K101
005	1.00	194001K102
017	1.00	194001K102
007	1.00	194001K103
006	1.00	194001K103
001	1.00	194001K105

PWG Total	17.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
020	1.00	001000G008
018	1.00	001000K403
019	1.00	001010A103
028	1.00	003000K103
048	1.00	003000K404
052	1.00	004020A305
031	1.00	004020A401
039	1.00	013000K113
049	1.00	013000K403
037	1.00	013000K411
043	1.00	014000G005
025	1.00	015000G001
057	1.00	015000K406
050	1.00	017000G007
036	1.00	026000A301
035	1.00	026000G004
027	1.00	059000A411
034	1.00	061000K407
051	1.00	072000K302

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
PS-I Total	19.00	

Group II

QUESTION	VALUE	KA
030	1.00	002000G005
033	1.00	006000K410
022	1.00	010000A302
023	1.00	010000A403
029	1.00	010000K103
055	1.00	011000A101
044	1.00	012000K201
026	1.00	012000K604
021	1.00	016000K101
054	1.00	028000K601
056	1.00	034000G001
042	1.00	035010K503
045	1.00	062000K410
038	1.00	064000A307
047	1.00	064000G007
046	1.00	064000G010
040	1.00	086000A202
PS-II Total	17.00	

Group III

QUESTION	VALUE	KA
032	1.00	005000K408
024	1.00	041020A102
041	1.00	041020K302
053	1.00	078000K303
PS-III Total	4.00	
PS Total	40.00	

EMERGENCY PLANT EVOLUTIONS

Group I

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
059	1.00	000005A101
058	1.00	000005A203
060	1.00	000005G005
062	1.00	000011A113
061	1.00	000011A210
063	1.00	000015G007
064	1.00	000026G012
065	1.00	000029G010
066	1.00	000029K312
068	1.00	000040A204
067	1.00	000040K304
069	1.00	000051A202
071	1.00	000055A106
070	1.00	000055G011
072	1.00	000057A106
073	1.00	000057A203
074	1.00	000059A205
075	1.00	000067A108
076	1.00	000068A121
077	1.00	000068K201
078	1.00	000068K312
079	1.00	000069K301
081	1.00	000074K103
080	1.00	000074K201

EPE-I Total	24.00	

Group II

QUESTION	VALUE	KA
083	1.00	000007A104
084	1.00	000007A206
082	1.00	000007K106
085	1.00	000008K101
086	1.00	000008K301
088	1.00	000009A239
087	1.00	000009K321
089	1.00	000022A202
090	1.00	000022K302
092	1.00	000025K202
091	1.00	000025K301
093	1.00	000032G003
094	1.00	000037A212
095	1.00	000038K305

S R O E x a m P W R R e a c t o r

O r g a n i z e d b y K A G r o u p

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
096	1.00	000054G010
097	1.00	000065A104
EPE-II Total	16.00	

Group III

QUESTION	VALUE	KA
099	1.00	000028A202
098	1.00	000036G007
100	1.00	000056G011
EPE-III Total	3.00	
EPE Total	43.00	
Test Total	100.00	

SALEM UNIT 1/2 OPERATIONS

INFO ONLY

PROCEDURE NUMBER: S2.OP-DD.ZZ-OD74(Z) *REV.* 2

PROCEDURE NAME
TANK AND SYSTEM CAPACITY CURVES

SPONSOR ORGANIZATION: SALEM OPERATIONS

USE CATEGORY: II

REVISION SUMMARY:

- This is a Limited Revision. Only SECTION 1 Table of Contents, and Page 57 are affected by this revision. All other pages remain unaffected and continue as Rev. 1
- Only affected pages are being redistributed by this revision.
- Added page 57 for a new curve supplied by Reactor Engineering titled "Reactor Cavity Volume vs. Elevation". Ref: Engineering Memo 93066 dated 4-27-93.

REFERENCES: S-C-VAR-0326-1,2,3

ANCILLARY INFORMATION:

- Tank Curves were created by the Nuclear Engineering Sciences Department.
- Tank Sketches were created by the Salem Operations Department.
- System Operating Curves were created, approved for use, and issued by the System Planning and System Operations Department.

IMPLEMENTATION REQUIREMENTS:
None Required.

APPROVED: _____ *DATE:* 12-10-93
OPERATIONS MANAGER - SALEM

TABLE OF CONTENTS
TANK AND SYSTEM CAPACITY CURVES
UNIT 2

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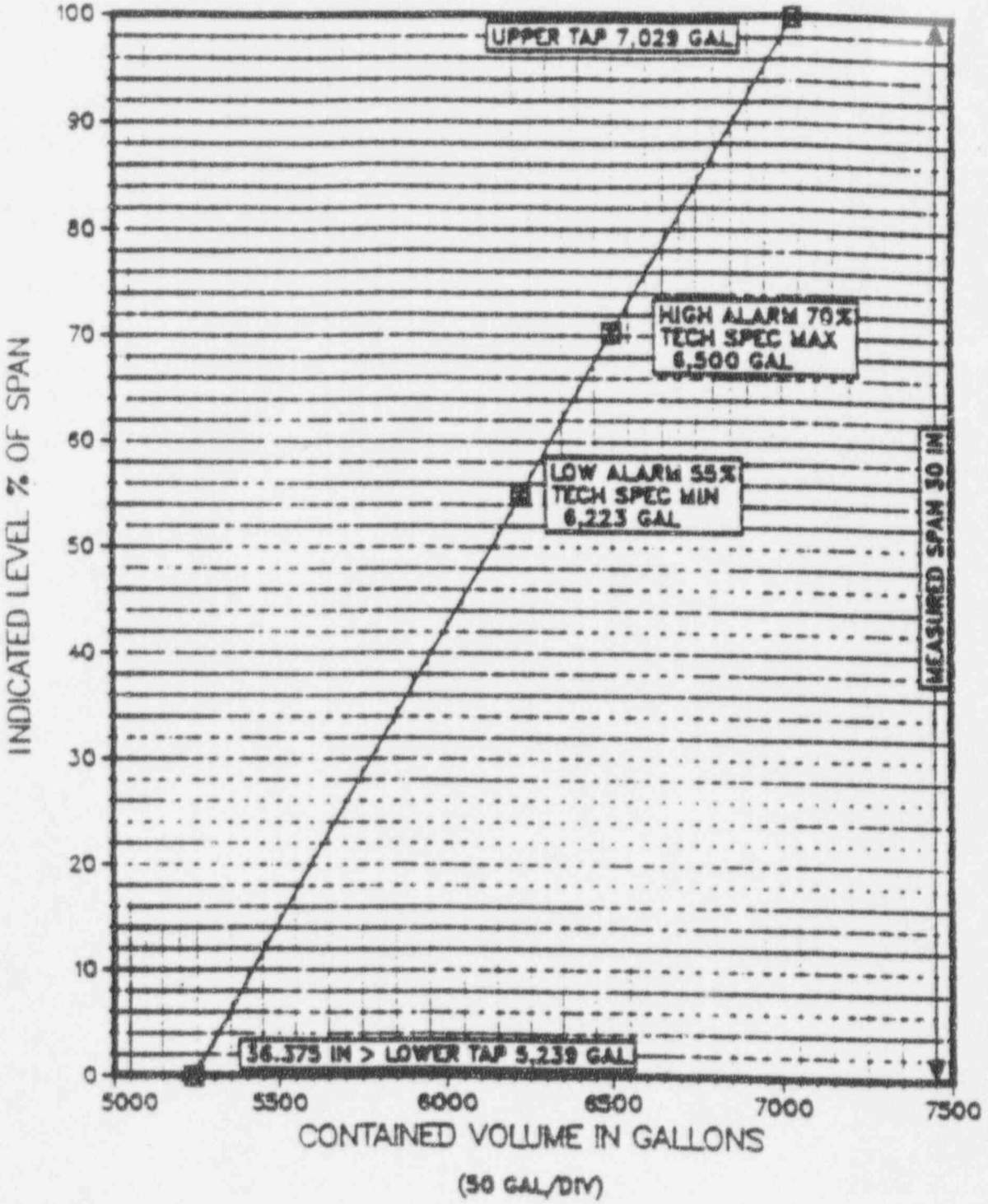
TANK VOLUME CURVES AND SKETCHES
UNIT 2

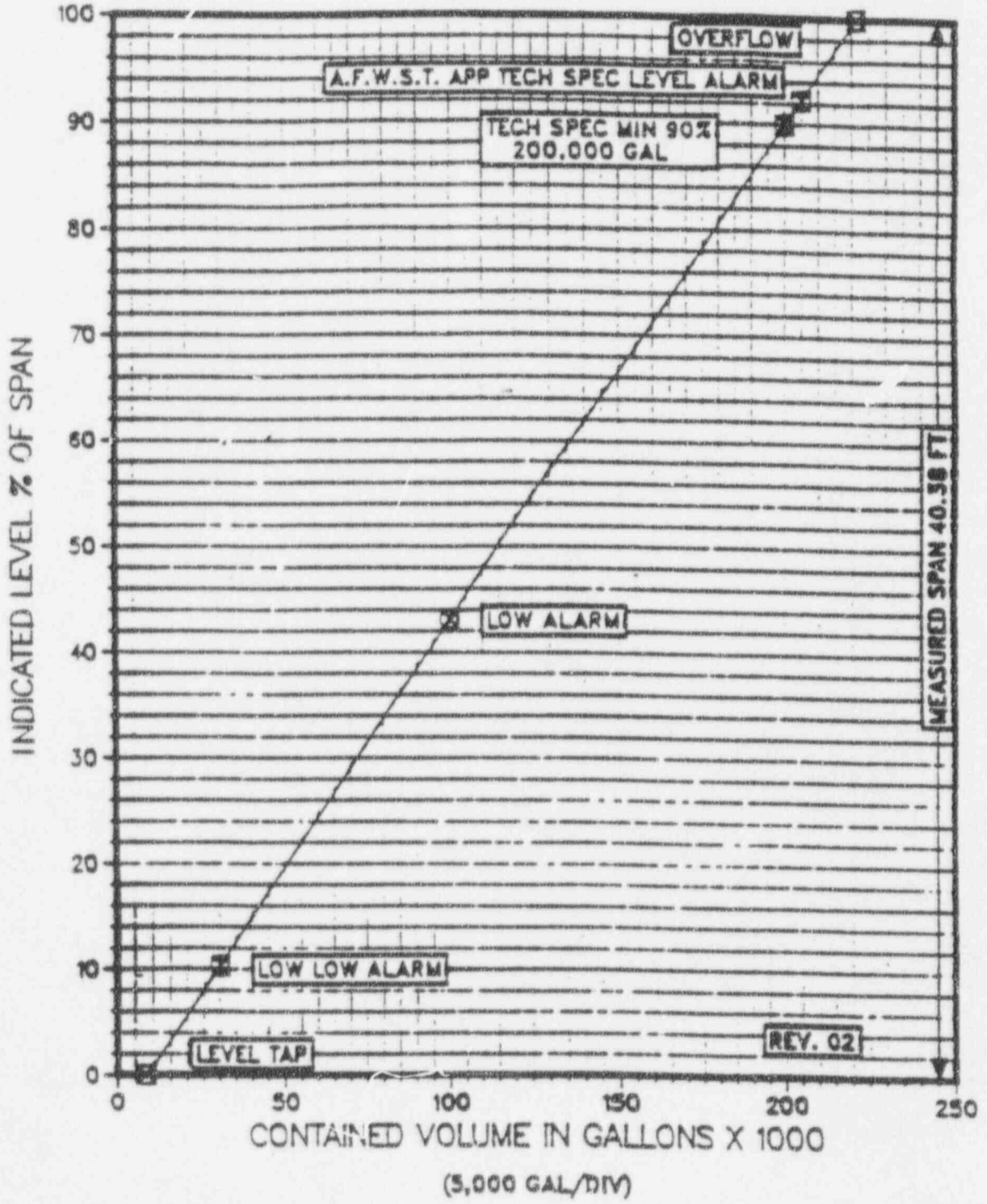
TABLE OF CONTENTS - SECTION 1
TANK VOLUME CURVES AND SKETCHES
UNIT 2

TABLE OF CONTENTS - SECTION 1
TANK VOLUME CURVES AND SKETCHES
UNIT 2

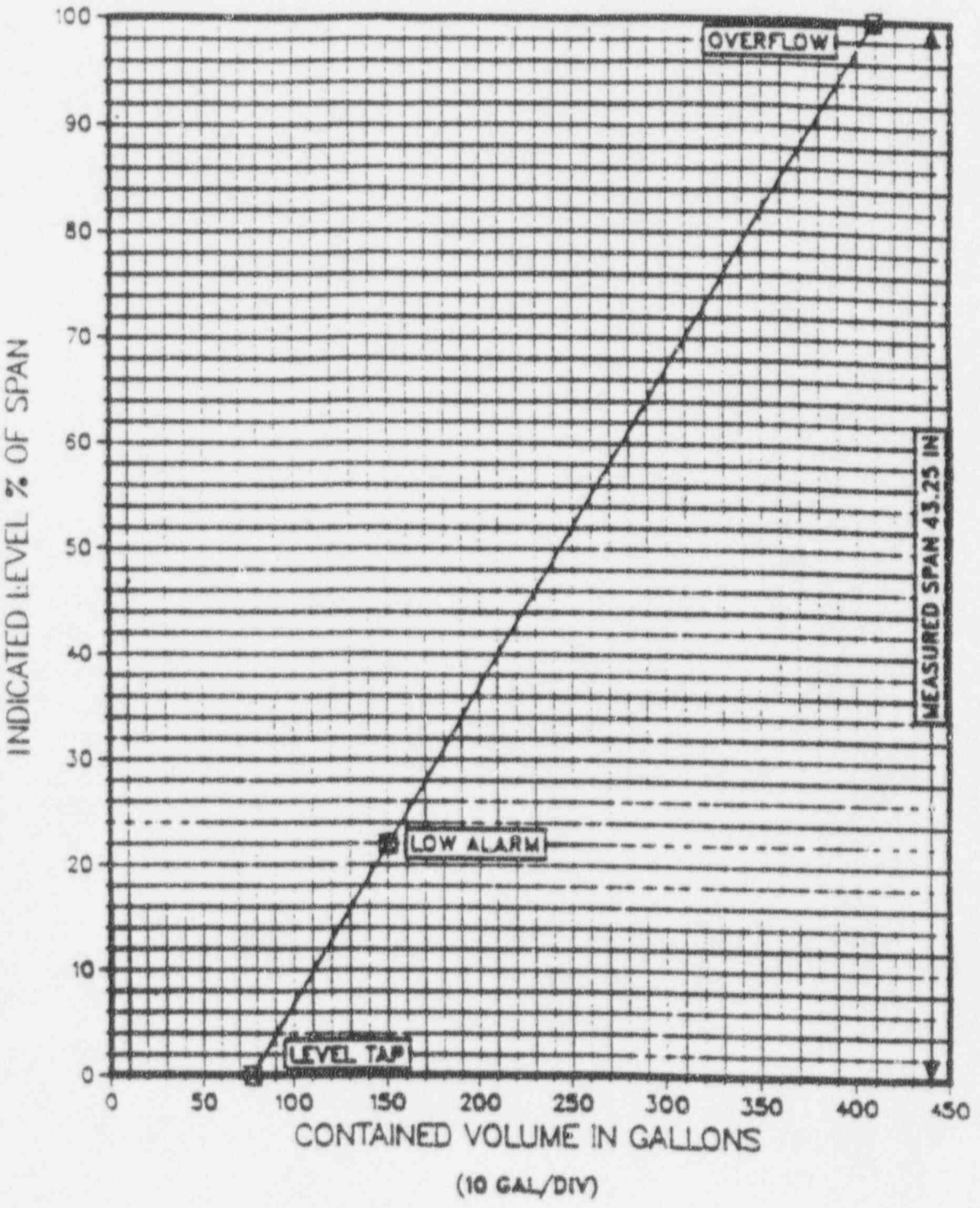
UNIT 2 ACCUMULATORS



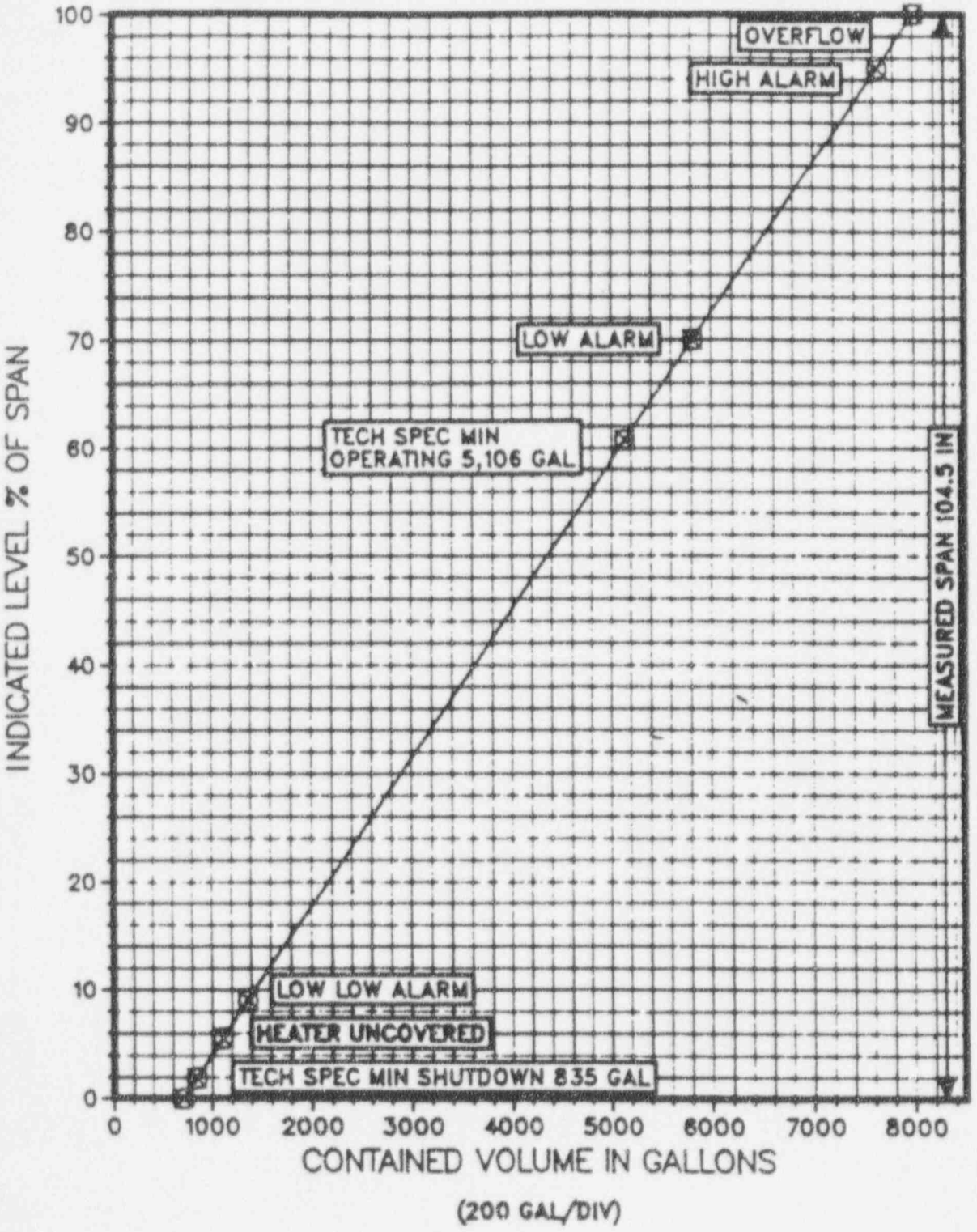
AUXILIARY FEEDWATER STORAGE TANK



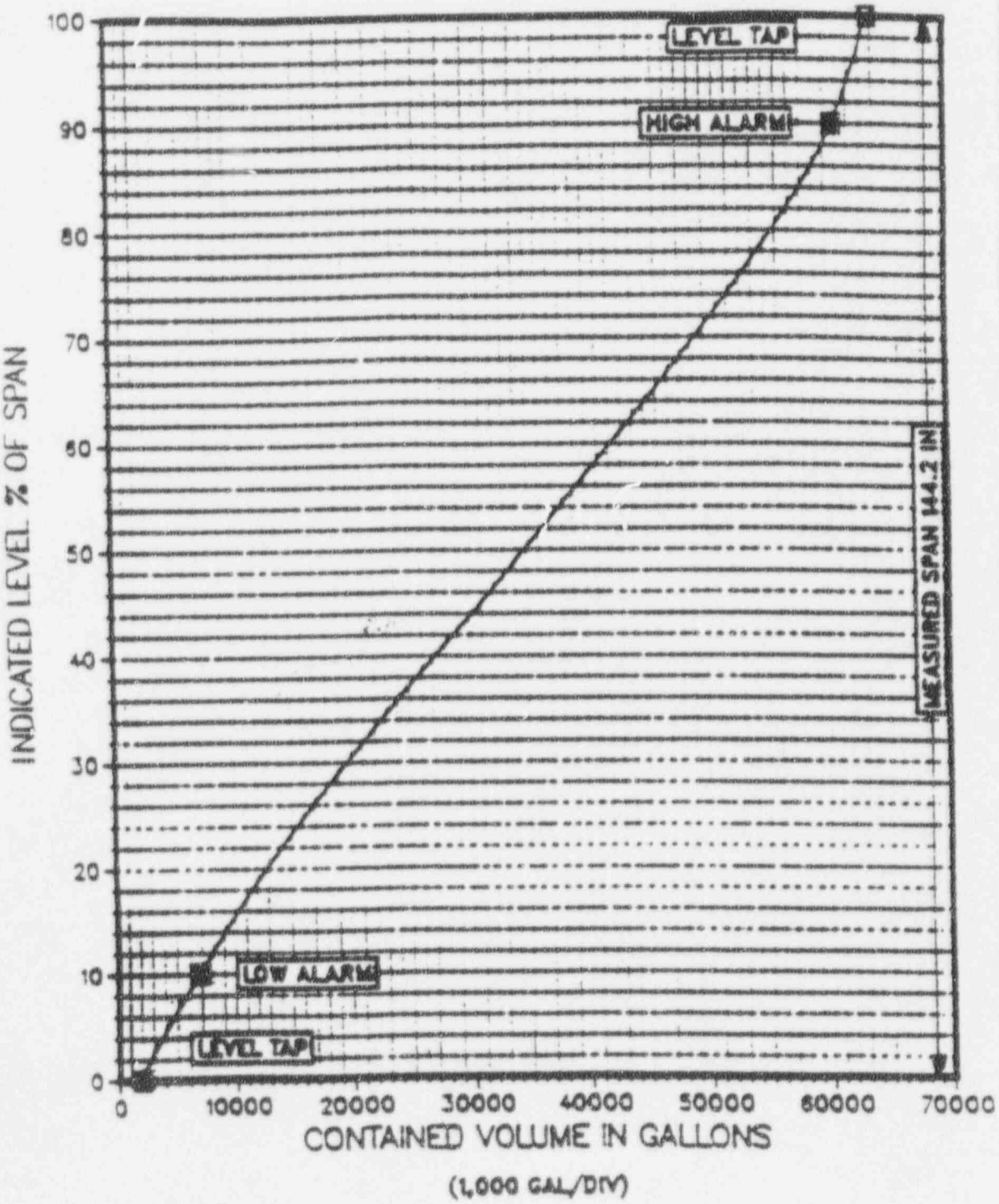
BORIC ACID BATCH TANK



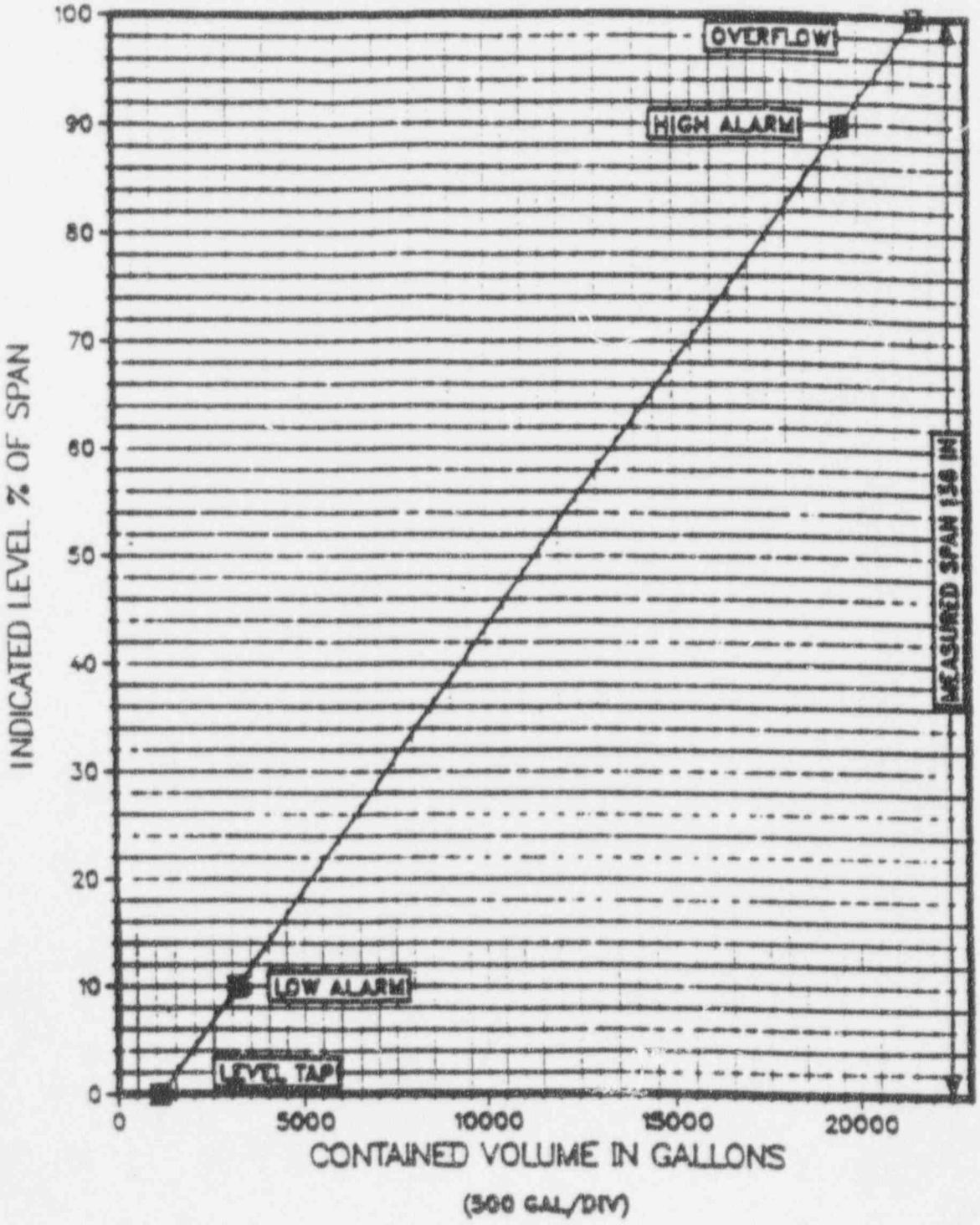
UNIT 2 BORIC ACID STORAGE TANK



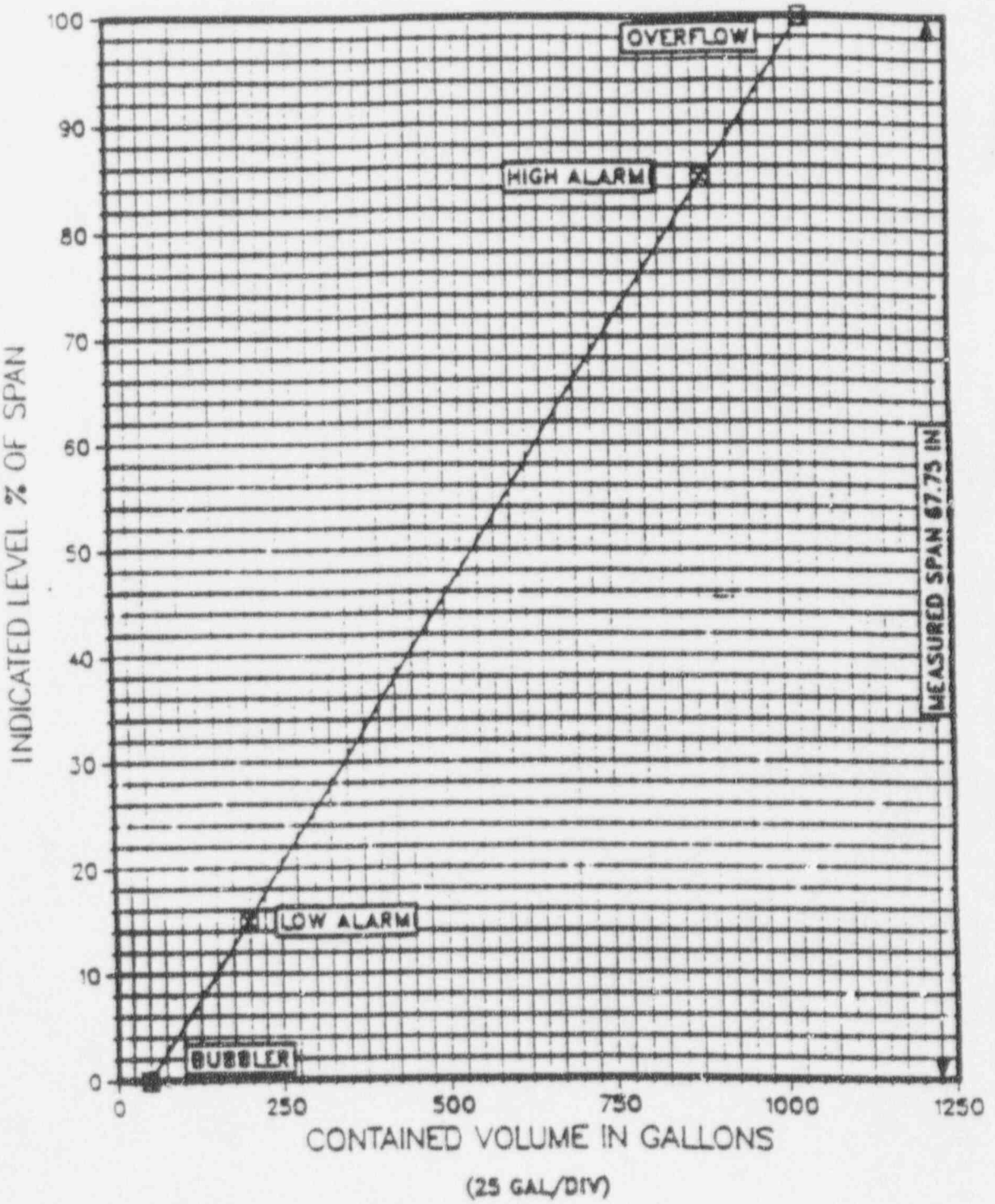
CHEMICAL & VOLUME CONTROL HOLD-UP TANK



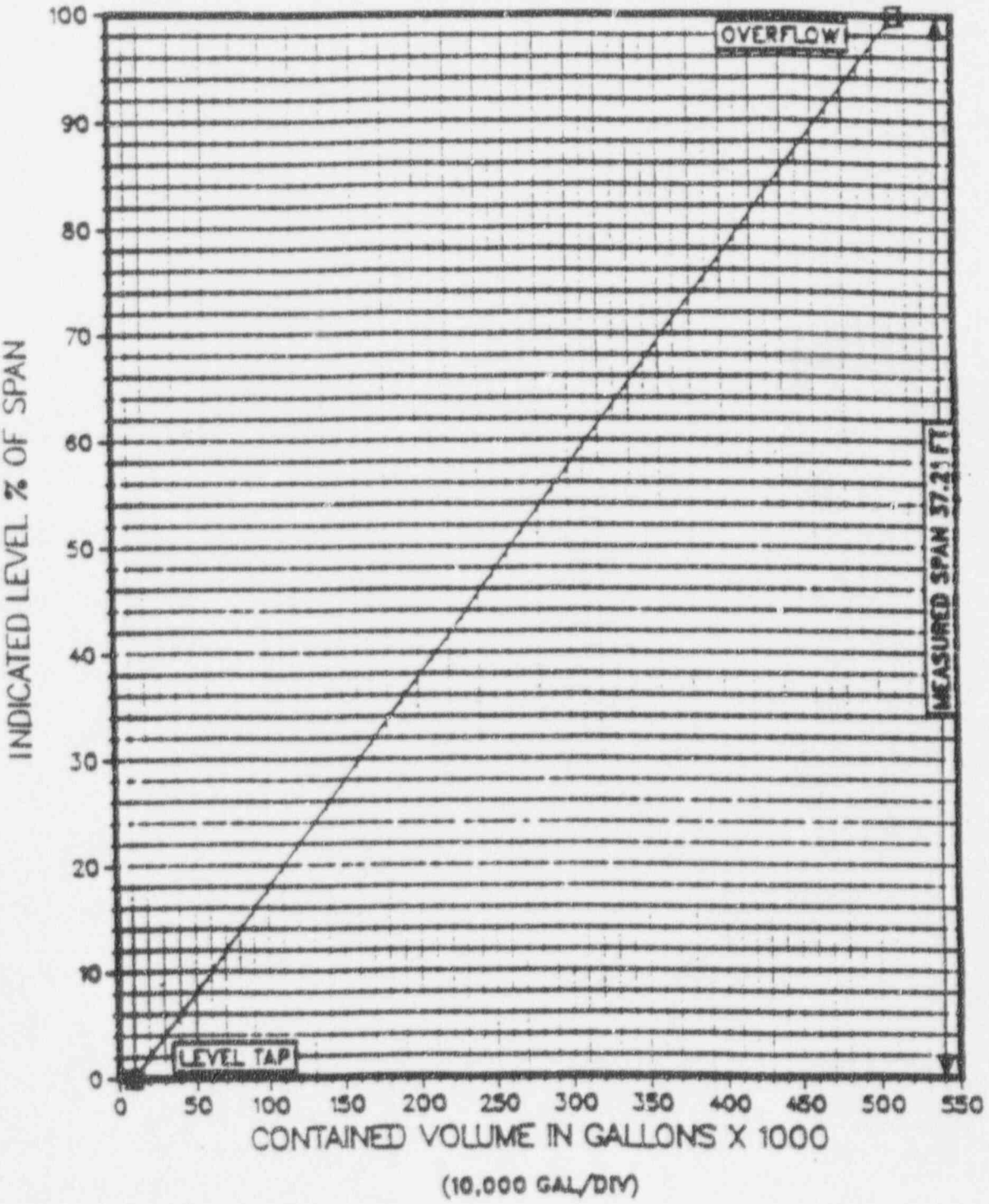
CHEMICAL & VOLUME CONTROL
MONITOR TANK



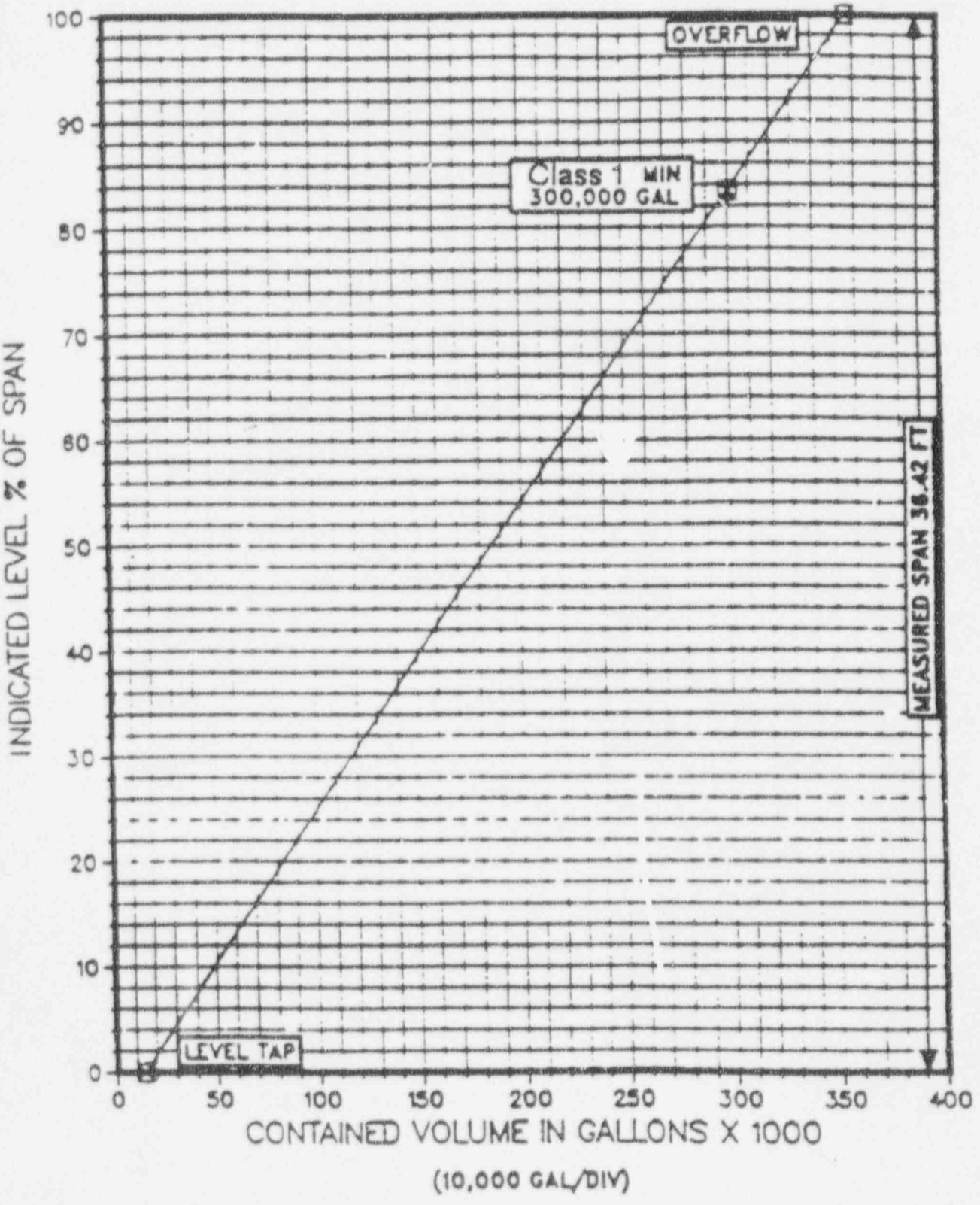
CONCENTRATES HOLDING TANK



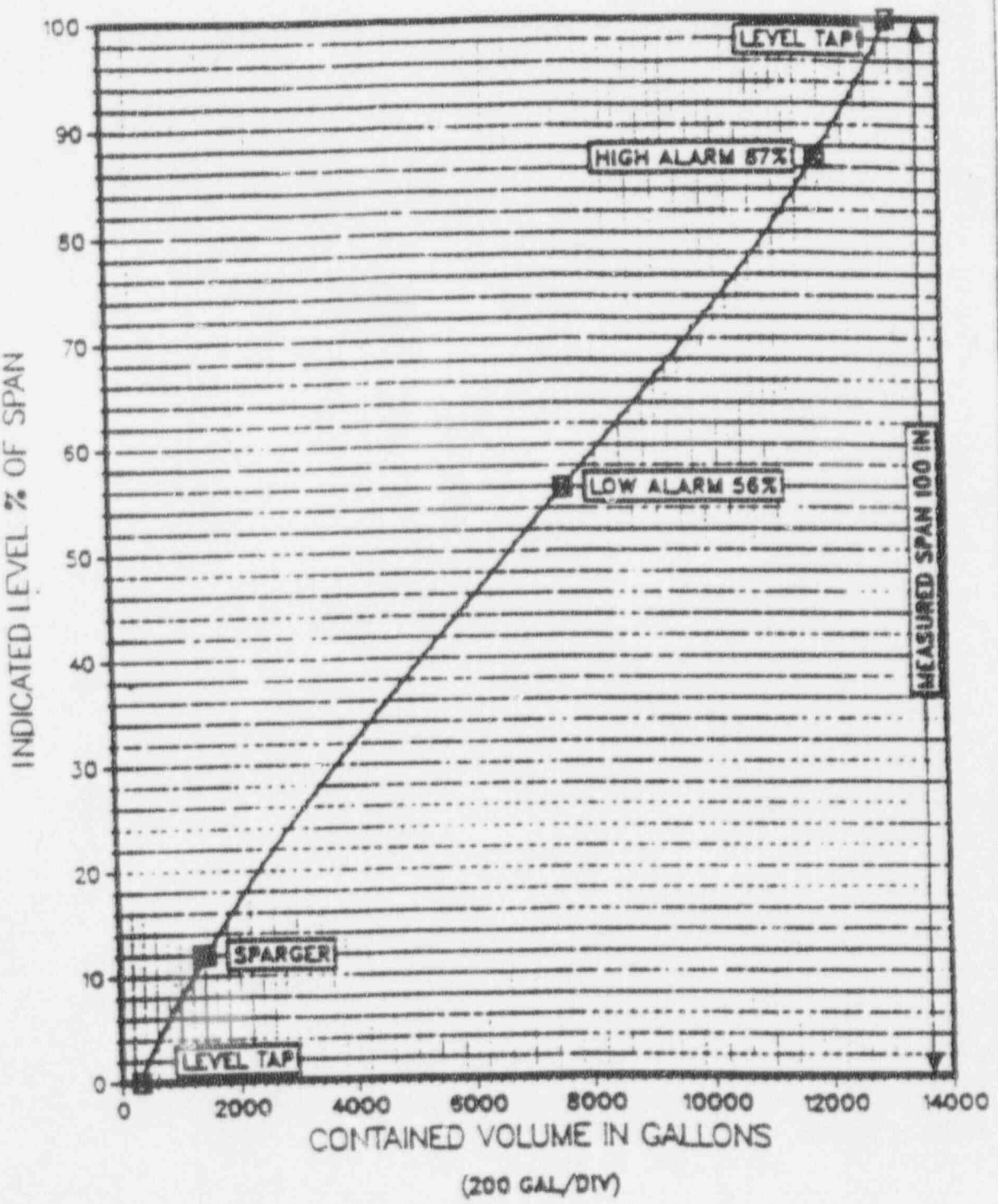
DEMINERALIZED WATER STORAGE TANK



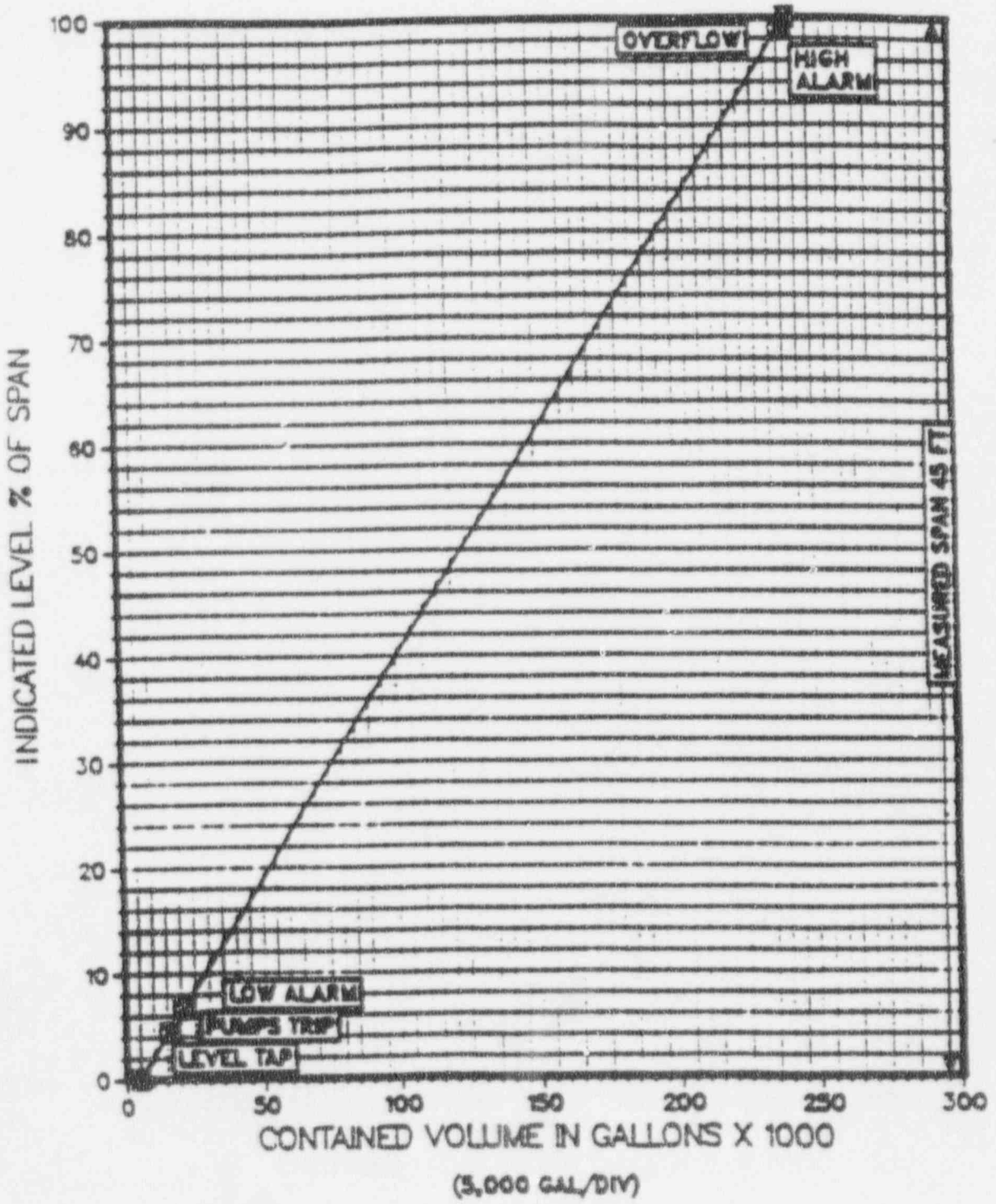
FRESH WATER & FIRE PROTECTION STORAGE TANK



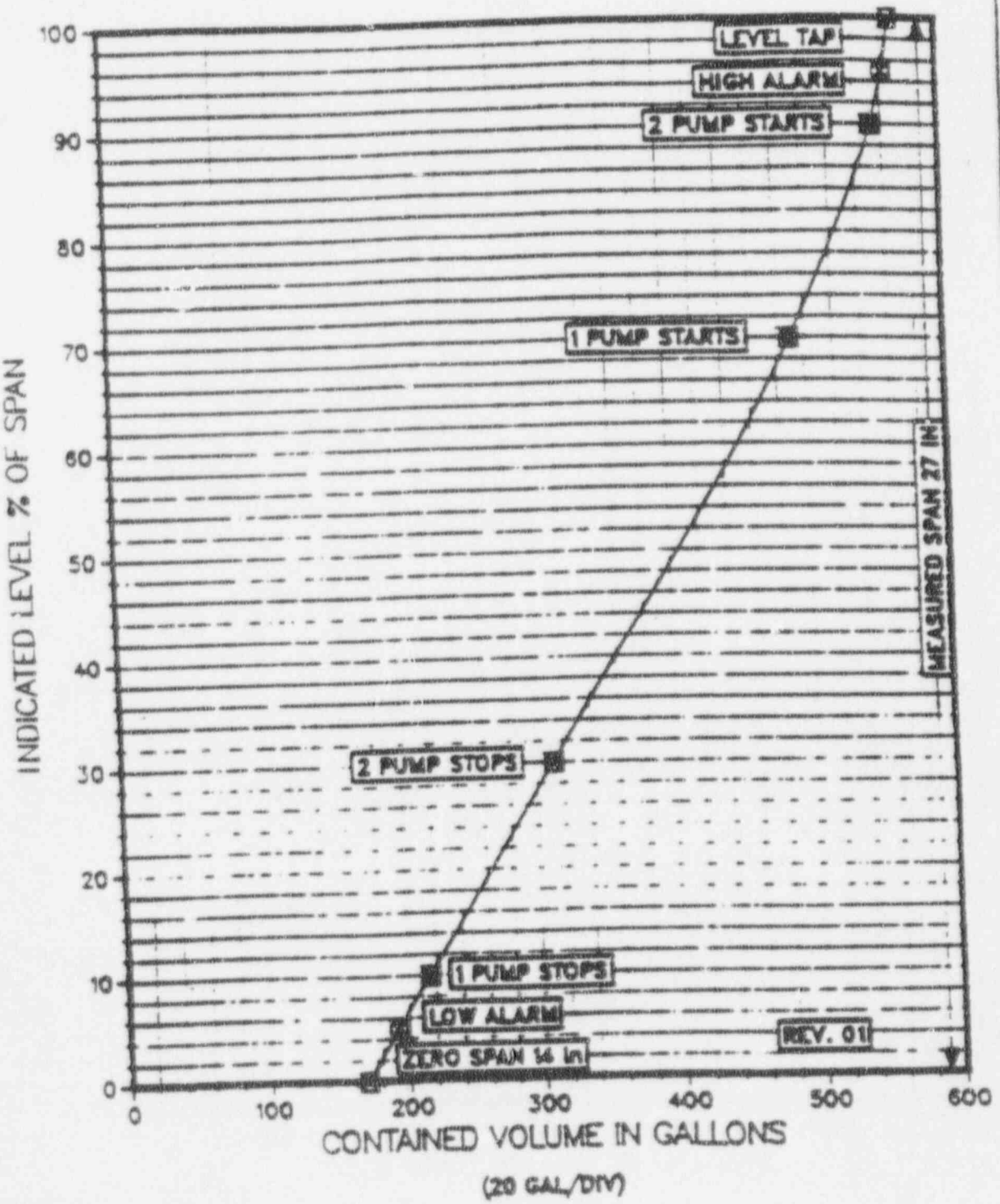
PRESSURIZER RELIEF TANK



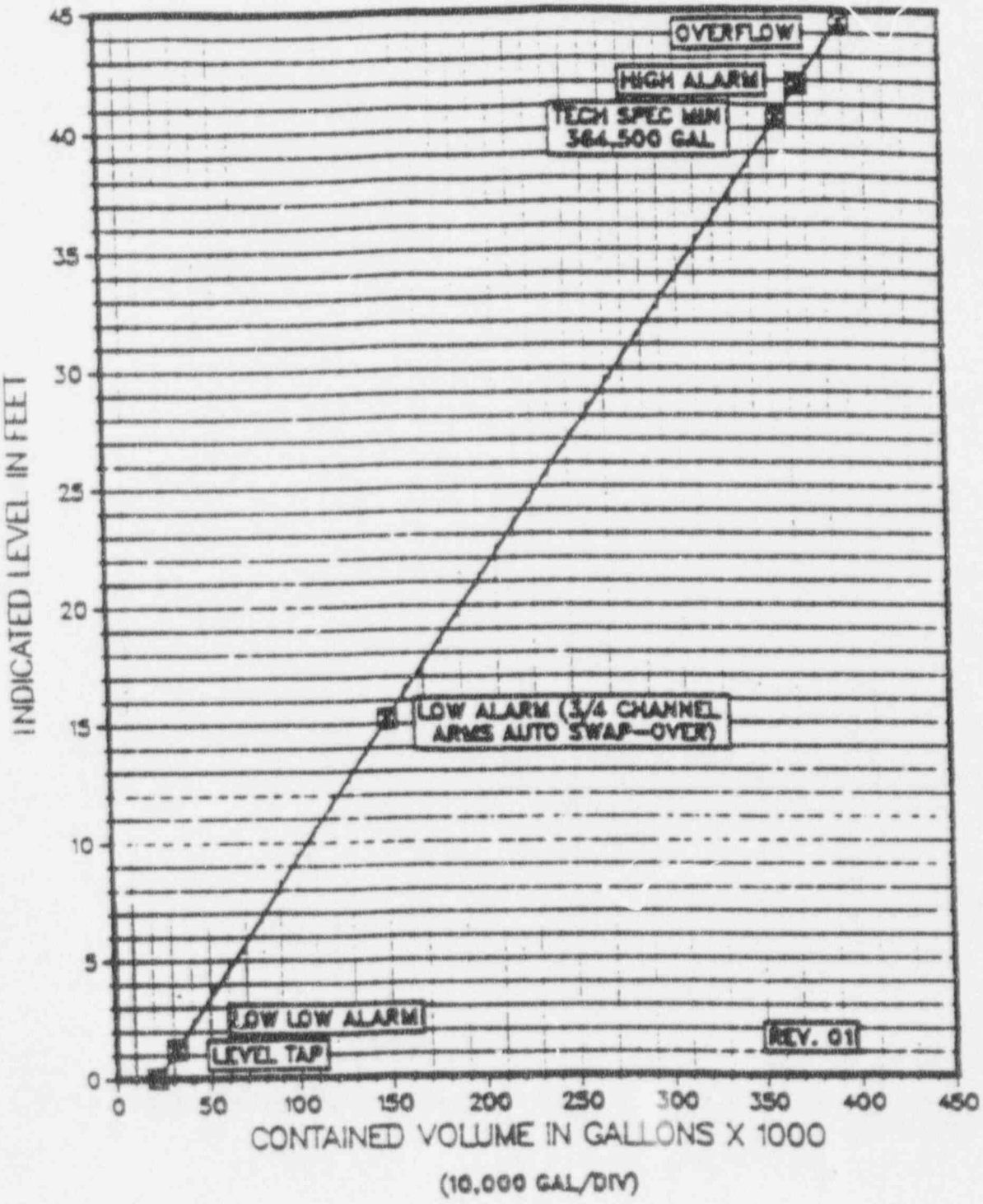
PRIMARY WATER STORAGE TANK



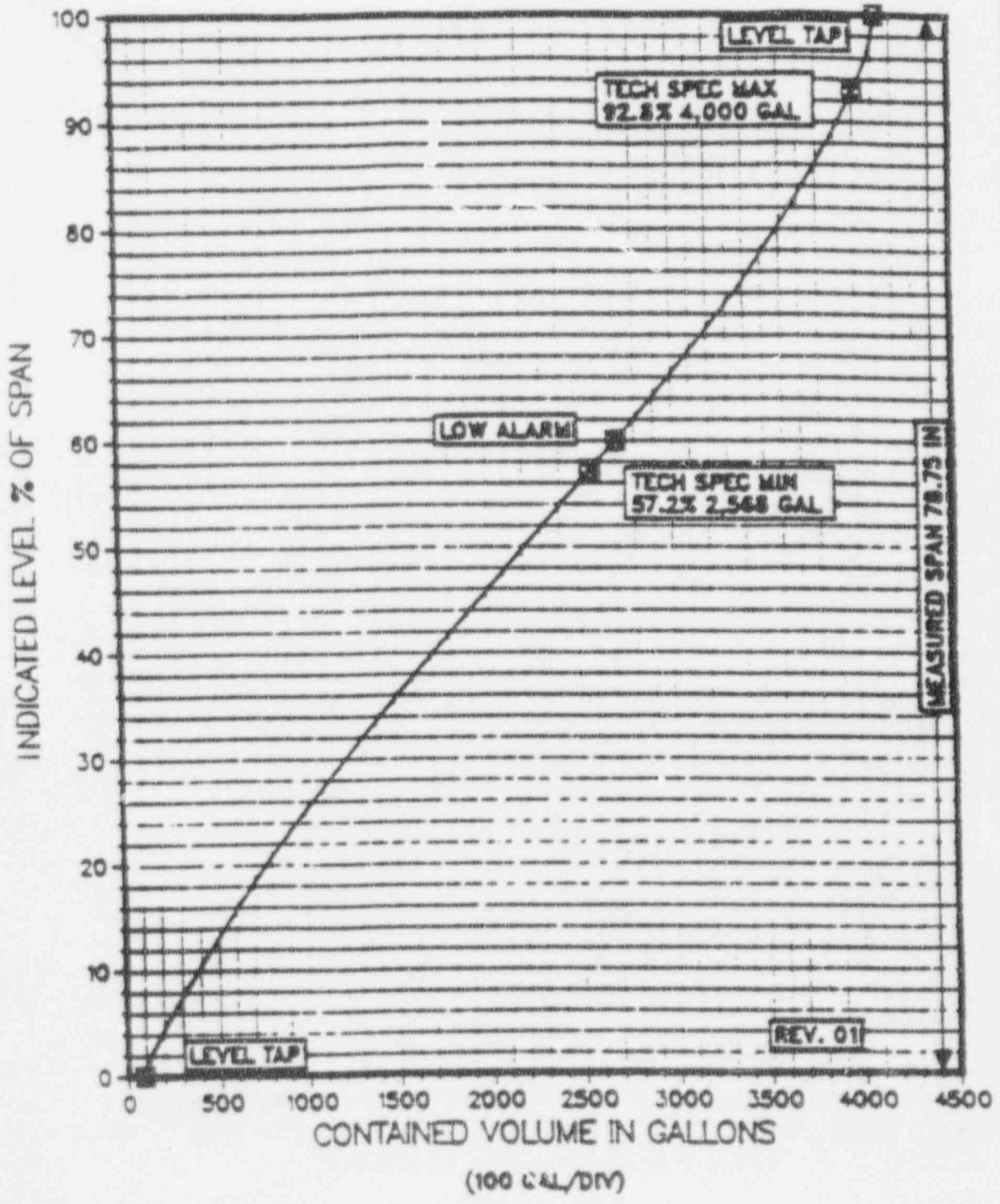
REACTOR COOLANT DRAIN TANK



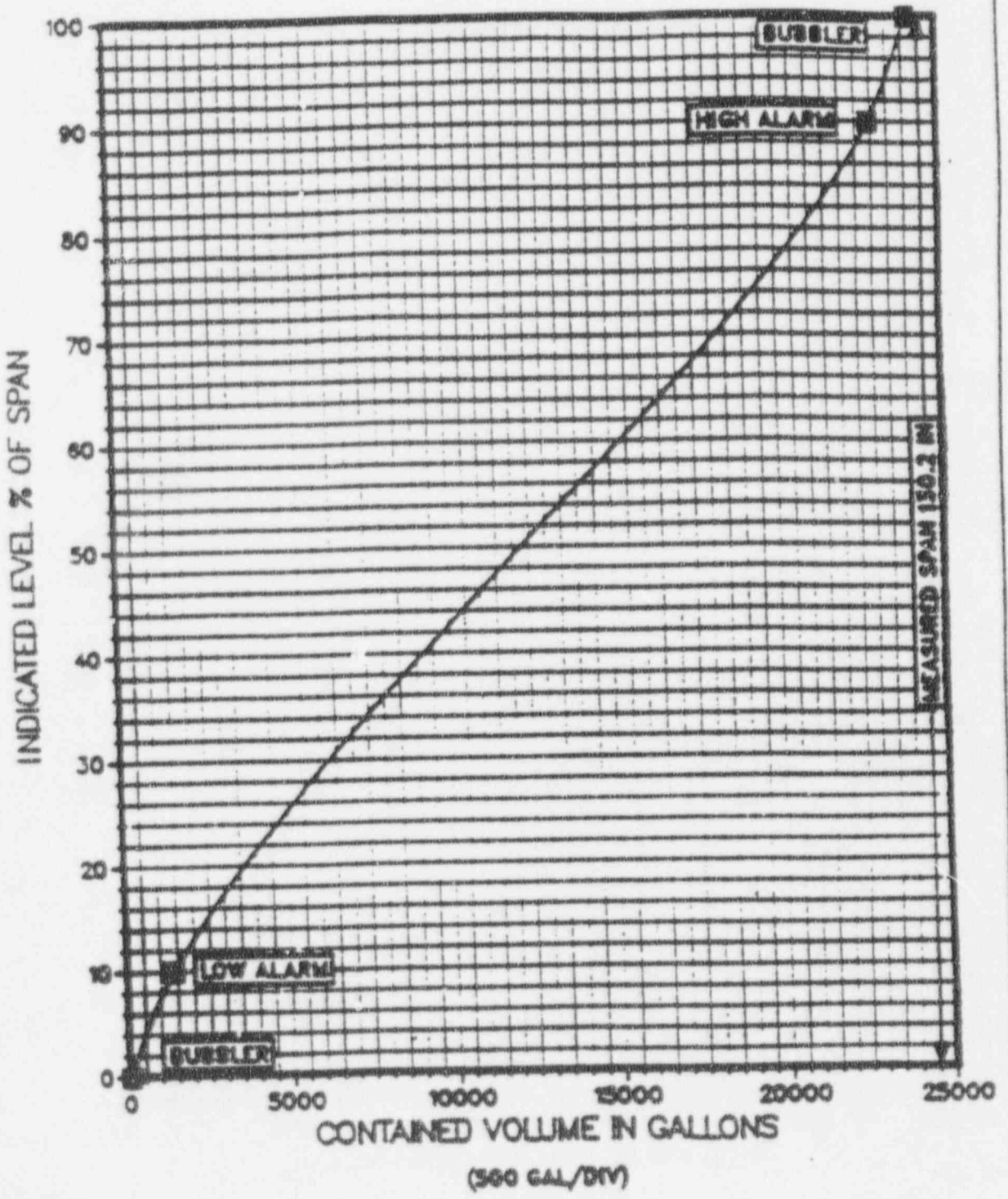
UNIT 2 REFUELING WATER STORAGE TANK



SPRAY ADDITIVE TANK



WASTE HOLD-UP TANKS & WASTE MONITOR HOLD-UP TANK



ATTACHMENT 3

SIMULATION FACILITY REPORT

Facility: Salem Units 1 & 2

Docket Nos: 50-272 & 311

Operating Tests Administered from: June 21-24, 1994

This form is used only to report simulator observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

The simulator exhibited one major breakdown during the examination week, at which time the scheduled simulator scenarios had to be rescheduled until the end of the week. During this breakdown, it was necessary to enlist the services of a tech rep, since all previous measures to locate and isolate the cause of the failure had been to no avail. The problem was eventually isolated to a pair of grounded I/O wires in a room across the hallway from the simulator. This room contained the beginnings of a future simulated remote shutdown panel. Once these wires were isolated, the random I/O problem disappeared. Also, a 5-volt power supply was replaced due to borderline readings.

One simulator fidelity issue was identified during the conduct of the facility's annual dynamic simulator examination, when wide-range steam generator level did not track during a steam generator level transient. The failure had been previously identified by the facility, a deficiency report had been written, but troubleshooting had not resolved the discrepancy at the time of the requalification examination. Because tracking of wide-range level is only occasionally erratic, no compensatory measures were instituted by the facility during the requalification examination.