ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/97-01
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	January 27 through February 14, 1997
Inspector:	T. O. McKernon, Chief Examiner
Approved By:	J. L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: Simulation Facility Report

Attachment 3: Facility Post Examination Comments and NRC Review

Attachment 4: Final Written Examination and Answer Key

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EXECUTIVE SUMMARY

River Bend Station
NRC Inspection Report 50-458/97-01

This inspection evaluated the competency of a senior operator license applicant for issuance of an operating license at the River Bend Station facility. The initial license examination was developed by the facility using the pilot process program guidance contained in Generic Letter 95-06 and NUREG-1021, Revision 7 and Supplement 1, "Operating Licensing Examiners Standards." An NRC examiner reviewed, approved, and administered the examination. The initial written examination was administered to the applicant on January 27, 1997 by both facility and NRC proctors. The operating test was administered on January 31, 1997 and March 12, 1997. The senior operator applicant displayed the requisite knowledge and skills to satisfy the requirements of 10 CFR 55 and was issued the appropriate license.

Operations

- The draft examination outlines as submitted were not adequate for administration. The operating test control room systems proposed were of narrower scope than allowed by NUREG-1021 (Section O5.1.1).

- Two potential examination compromises occurred, which were contrary to examination security practices agreed to by licensee staff (Section O5.1.2).

- The applicant passed all portions of the initial license examination which satisfied the requirements of 10 CFR 55.33(a)(2), and was issued the appropriate license in accordance with 10 CFR 55.51. However, the applicant performed marginally on the written examination (Sections O4.1, O4.2).

- The licensed operator requalification program performance was reviewed to ascertain operator performance. The licensee's licensed operators performed acceptably during recent licensed operator training cycle examinations (Section O5.3).

- The licensee's written examination prereview was not of sufficient quality as evidenced by over 5 percent of the written examination being identified for change in the postexamination review (Section O5.1.2).

Engineering

- The Updated Final Safety Analysis Report wording was inconsistent with that of the Improved Technical Specification and the observed plant practices related to qualification requirements for licensed senior operators (Section E2.1).

Report Details

Summary of Plant Status

The unit remained at 100 percent power during the inspection period. No significant plant challenges to the operating crew were experienced.

I. Operations

O4 Operator Knowledge and Performance

O4.1 Initial Written Examination

a. Inspection Scope

The written examination was co-administered by the licensee training representative and the NRC examiner onsite on January 27,1997, in accordance with guidance provided by the chief examiner. This action was taken because of a potential compromise of examination security which occurred the week prior to the written examination week. The NRC examiner graded the written examination and the licensee staff performed a postexamination analysis. As a result of the licensee's post examination analysis, postexamination comments were submitted to the NRC on March 3, 1997.

b. Observations and Findings

The applicant passed the written examination with a score of 80.4 percent. After final NRC review it was determined that the applicant missed 19 out of 97 questions. The missed questions were reviewed by both the licensee and the NRC to ascertain whether a generic weakness or training deficiency existed. No specific generic weakness was identified. However, the postexamination review determined that the applicant demonstrated a potentially weak overall system knowledge in that questions related to detailed systems knowledge were typically missed. As such, the applicant performed marginally on the examination.

c. Conclusions

The applicant passed the written examination portion of the license evaluation. Evaluation of the applicant's performance indicated a potential weakness in the depth of the applicants overall system knowledge.

O4.2 Initial Operating Test

a. Inspection Scope

The NRC examiner administered the various portions of the operating test to the applicant on January 31 and February 12, 1997. The applicant participated in two dynamic simulator scenarios; one while acting as the reactor operator at-the-controls and acting as the control room supervisor in the second. The applicant also received a walkthrough test which consisted of ten system and four administrative area tasks and questions.

b. Observations and Findings

During the simulator portion of the examination, communications practices observed in the simulator were good. Operator performance in the simulator was also good. The applicant passed this portion of the examination and demonstrated good command and control skills.

During the job performance measure portion of the examination, the applicant demonstrated good plant knowledge and a familiarity in locating components, support items, and reference material. The administrative portion of the examination was also administered during the plant walkthroughs. The applicant demonstrated a high degree of familiarity with the plant's administrative functions and requirements. The applicant passed these portions of the examination.

c. Conclusion

The applicant passed the operating test portion of the examination. Good communication skills as well as command and control skills were observed. The applicant demonstrated a high degree of familiarity with plant procedures and in locating components in the plant.

O5 Operator Training and Qualification

O5.1 Initial Licensing Examination Development

The facility licensee developed the initial licensing examination in accordance with guidance provided in Generic Letter 95-06, "Changes in the Operator Licensing Program."

O5.1.1 Examination Outline

a. Scope

The licensee submitted the initial examination outlines on November 21 and December 10, 1996. The chief examiner reviewed the submittals against the requirements of NUREG-1021, "Licensed Operator Examiner Standards" Revision 7, Supplement 1, and NUREG/BR-0122, "Examiner's Handbook for Developing Operator Licensing Written Examinations," Revision 5.

b. Observations and Findings

The chief examiner determined that the initial examination outlines generally satisfied the above requirements. However, minor changes to the written examination outline were made by revising a knowledge/ability designator. The chief examiner also noted that the job performance measures outline did not meet the requirements of Examiner Standard 201-3, Item 3b, in that safety functions for Job Performance Measures 4 and 8 were repeated which resulted in the job performance measures not covering at least seven different safety functions. Other observations by the examiner were minor. The comments on the outlines were discussed with the licensee author and revisions were made and submitted along with the draft initial examination.

c. Conclusion

The draft examination outlines as submitted were not adequate for administration. The operating test control room systems proposed were of narrower scope than allowed by NUREG-1021.

O5.1.2 Examination Package

a. Inspection Scope

The licensee submitted the completed draft examination package between December 10, 1996 and January 6, 1997. The chief examiner reviewed the submittal against the requirements of NUREG-1021, "Licensed Operator Examiner Standards" Revision 7, Supplement 1 and NUREG/BR-0122, "Examiner's Handbook for Developing Operator Licensing Written Examinations," Revision 5.

b. Observations and Findings

The draft written examination was transmitted by the licensee to the NRC in a letter dated December 10, 1996. The draft written examination was considered properly discriminating. It was responsive to a knowledge and abilities sample plan submitted by a letter dated November 21, 1996, which was approved by the chief examiner. The chief examiner approved the draft examination based upon a review of reference material provided with each question. Few substantive comments were made with most comments of an editorial nature. The written examination was considered adequate as submitted. In response to chief examiner comments, the licensee enhanced questions 35, 43, 55, 56, 62, 69, 70, 76, 77, 78, 81, 88, 90, 94, and 97.

Additionally, due to a potential examination compromise, other questions were replaced. The compromise occurred when a licensed operator, with examination content knowledge and on the security agreement, interacted with the applicant to complete systems qualification the week before the examination. The control room supervisor who was also on the examination security agreement observed the operator and applicant and halted the activity in progress. The licensee initiated a condition report, conducted investigations and evaluated the draft examination for possible overlap with the system qualification job task evaluations. As a result, the licensee replaced questions 19 and 33 on the written examination. There was no indication of an actual compromise to the examination.

The licensee provided postexamination comments on seven written questions which are attached to this report (Attachment 3). Review of the licensee's submitted post examination comments resulted in accepting licensee recommendations in deleting three questions (34, 41, 59); a change in the correct answer to one question (15); and the acceptance of two correct answers for another question (46). The review also determined that the licensee's recommendations for questions 80 and 89 were not valid and the questions were retained. Additionally, the NRC reviewed other questions and determined that question 27 had two possible correct answers. As a result, after incorporating these changes, the applicant received a score of 78 out of 97, equivalent to 80.4 percent. The as-given written examination and answer key are attached to this report (Attachment 4).

Because postexamination reviews identified changes based on technical accuracy of over 5 percent of the written examination, the licensee was asked to explain why their staff had failed to identify these issues before the examination administration and to describe corrective actions planned to assure that the next licensed operator written examination is of higher quality.

The licensee submitted three dynamic scenarios, including one backup scenario which was not used during the examination. The chief examiner had few comments related to the proposed scenarios. Some general comments related to statements for operator actions being explicit were made. The second scenario did not meet the criteria of Examiner Standard 604-1 for contingency actions after emergency operator performance entry. Other minor spelling errors were also noted. The licensee reviewed the comments and made the necessary revisions prior to onsite validation the week of January 27, 1997.

During the week of January 27, 1997, a second potential examination compromise occurred when a licensed operator involved in simulator scenario validation inadvertently left some rough notes related to the first scenario in the simulator. The notes were discovered immediately afterward by the requalification crew entering the simulator for requalification training. The licensee took prompt actions to enter the requalification crew on the security agreement, brief all individuals on the security agreement about the incident and the need for continued vigilance since the examination was postponed. While the second incident was unrelated, the two events caused the licensee staff to reevaluate the adequacy of their security measures and to take both short and long-term corrective actions. The licensee replaced the entire set of simulator scenarios, including the backup scenario, to ensure that no overlap existed with the potentially compromised scenario. The dynamic simulator portion of the examination was postponed until the week of February 10, 1997.

The chief examiner validated the replacement scenarios during the week of February 10, 1997. The examiner verified that the replacement scenarios contained no overlap with any of the removed scenarios. The examiner also noted that the replacement scenarios required modification before administering to ensure the examiner standard guidances were fully met with regard to Examiner Standard 301-5, "Transient and Event Checklist."

The licensee also submitted both job performance measures and questions to cover the administrative section of the walkthrough test. The chief examiner reviewed the job performance measures and noted a number of comments which were discussed informally with the licensee examination author. Most of the comments related to the adequacy of the followup questions (e.g., multipart answers required, whether question is open or closed reference). Some other comments resulted in the job performance measure being replaced. For example, job performance measure "Manually Startup Standby Gas Treatment Train A taking a suction on Outside Air" was considered too simplistic in that the entire job task only had two active steps performed by the applicant. In another instance, a job performance measure had potential overlap with an event in one of the scenarios. The licensee was very responsive in reviewing the comments and making the necessary revisions.

c. Conclusions

Overall, the written examination and operating test materials submitted were of fair quality and discriminated at the appropriate license level. However, because of NRC comments on the initial draft examinations and potential security compromises, a number of revisions and revalidations prior to examination administration were required. Additionally, because the licensee submitted over 5 percent of the written examination for postexamination comment, the licensee's written examination prereview was not of sufficient quality.

O5.2 Simulation Facility Performance

a. Inspection Scope

The examiner observed simulator performance with regard to fidelity during the examination validation and administration.

b. Observations and Findings

It was observed that the simulator performed well during both the initial license examination validation and administration and during licensed operator requalification training conducted during the inspection period. Simulator fidelity issues were not observed during the examination and are reflected in Attachment 2.

c. Conclusions

The simulator supported the examination well. No simulator fidelity issues were observed.

05.3 Licensed Operator Requalification Program Status (IP 71001)

a. Scope

This portion of the inspection reviewed results from the past two licensed operator requalification training cycle examinations and conducted discussions with key operations training managers.

b. Observations and Findings

The inspector noted that no operating crew failures occurred during the September 2 through October 11, 1997, requalification training cycle. Further, during the annual requalification examinations conducted between October 14 and December 13, 1996, one crew out of nine failed the examination. It was noted that the crew failure during the annual examination was not one of the same two operating crews which had difficulty in the simulator during the past August 1996 cycle training. Additionally, the inspector discussed requalification training status with a key training manager and was informed that crews were taking greater ownership responsibility for their training to include better feedback and self-critiques of their simulator sessions.

c. Conclusion

The inspector concluded that operator performance based upon recent requalification cycle training and the annual requalification examination had been acceptable.

III. Engineering

E2 Engineering Support of Facilities & Equipment

E2.1 Review of the Updated Final Safety Analysis Report Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the Updated Final Safety Analysis Report descriptions. While preparing for the inspection discussed in this report, the inspector and the licensee reviewed the applicable portions of the Updated Final Safety Analysis Report that related to the areas inspected. The inspector noted that the Updated Final Safety Analysis Report training commitment was inconsistent with the observed plant practices, procedures, and/or parameters in that commitments stated in the licensee's Technical Specification. Specifically, Section TR 5.3, "Licensed Operator qualifications," of the Technical Specifications committed to the minimum qualifications of the supplemental requirements of NRC letter dated March 28, 1980. The Updated Safety Analysis Report, Section 13.2.1 commits individual training programs to comply with Regulatory Guide 1.8 and ANSI/ANS 3.1-1978. The licensee has implemented a systems approach to training program in accordance with Generic Letter 87-07. Further, the licensee's Procedure TPP-7-009, "Hot License Operator Training and Qualification Program," is designed to meet the requirements of ANS-3.1-1981 and Regulatory Guide 1.8. The licensee committed to revise the Final Updated Safety Analysis Report and Technical Specification to reflect their current training program commitments and requirements.

V. Management Meetings

X1 Exit Meeting Summary

The examiner presented the inspection results to members of the licensee management at the conclusion of the inspection on February 13, 1997. The licensee acknowledged the findings presented.

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

In addition, the licensee was informed by telephone on March 27, 1997, that a 30 day response request related to the quality of the licensee's written examination prereview would be included in the inspection report cover letter.

<u>ATTACHMENT 1</u>

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

D. Burnett, Senior Chemist
M. Chilson, Licensing Specialist
G. Davant, Licensing, Senior Staff
D. Dietzel, Initial Licensed Operator Training Representative
T. Gates, Plant Engineering Supervisor
T. Hildebrandt, Outage Manager
R. King, Director, Nuclear Safety and Regulatory Affairs
D. Lorfing, Licensing Supervisor
J. McGaha, Executive Vice President
J. McGhee, Operations Technical Assist
W. O'Malley, Operations Manager
A. Shahkarami, Engineering Manager
W. Stacey, Business Services Manager
C. Sutherland, Training Supervisor
W. Trudell, Operations Training Supervisor
M. Wagner, Supervisor, Initial Licensed Operator Training
L. Woods, Operations Supervisor
G. Zinke, Quality Assurance Manager

<u>NRC</u>

W. Smith, Senior Resident Inspector

ATTACHMENT 2

SIMULATION FACILITY REPORT

Facility Licensee: River Bend Station

Facility Docket: 50-458

Operating Examinations Administered at: RBS Training Center, St. Francisville, LA

Operating Examinations Administered on: January 31 through February 14, 1997

These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility, other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

NONE

ATTACHMENT 3

FACILITY POST EXAMINATION COMMENTS AND NRC POST EXAMINATION REVIEW

NRC Post-Examination Comment Review

Question #15; NRC disagrees. Accept only answer (c). The Normal Service Water System is a nonsafety-related system. The only TS surveillances which affect the system are containment isolation valve testing, IST pumps and valves program, and system radiation monitor functional testing. Therefore based upon the information provided in the stem only answer (c) is applicable.

Question #27; NRC post examination review. Accept either answer (a) or (b). Stem does not specify location of operator or the type of operator.

Question #34; NRC agrees. Delete question. Insufficient information provided in stem of question to indicate whether signal is sealed in or just spurious. This question was specifically identified during the prereview process, reviewed by the licensee and accepted in the as-administered format.

Question #41; NRC agrees, delete question.

Question #46; NRC agrees, accept either answer (a) or (b).

Question #59; NRC agrees, delete question.

Question #80; NRC disagrees. Keep as is. The question discriminates whether or not the applicant knows that with a loss of the "A" RPS Bus, the condenser air removal pumps are lost. As a result, main condenser vacuum will begin to decrease. The selection of (d) as an answer requires presumption that other leaks occur. While instrument air pressure will decrease in time, the plant response is protracted; the decrease in condenser vacuum begins almost immediately. Therefore, answer (b) is the most plausible answer based upon the conditions provided in the stem of the question.

Question #89; NRC disagrees. Keep as is. The question does not require the applicant to know the test criteria, but does require the applicant to understand where to look in order to call the rod "slow" or "inoperable". This question was accepted during the licensee's prereview process. Further, the training learning objective requires the student to identify the TS operability requirements for the CRD Hydraulic and CRD Mechanism systems.

Entergy

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61
P.O. Box 220
St. Francisville, LA 70775
Tel 504 381 4374
Fax 504 381 4872

John R. McGaha, Jr.
Vice President
Operations

March 3, 1997

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

Subject: River Bend Station Submittal of Formal NRC Examination Comments
River Bend Station - Unit 1
Docket No. 50-458

File No.: G9.5, G1.41.26

RBF1-97-0075
RBG-43761
RBEXEC-97-030

Ladies and Gentlemen:

In accordance with NUREG-1021, Revision 7, Supplement 1 (ES-402, Attachment 3), enclosed are comments on seven examination questions for your review and consideration. The enclosed comments and supporting documentation are provided as a result of the examination conducted at River Bend Station the week of January 27, 1997 and February 10, 1997. The chief examiner during the examination was Mr. Thomas McKernon.

If you have any questions, please contact Mr. Dave Dietzel at (504) 381-4246 or Mr. David G. Looney at (504) 381-3630.

Sincerely,

John R. McGaha, Jr.

John R. McGaha, Jr.
Vice President
Operations

JRM/kvm
enclosure

cc: U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Stop P1-37
Washington, DC 20555

NRC Sr. Resident Inspector
P. O. Box 1050
St. Francisville, LA 70775

David Wigginton
NRR Project Manager
U. S. Nuclear Regulatory Commission
MS OWFN 13-H-3
Washington, DC 20555

Mr. Thomas McKernon
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mr. John L. Pellet
Chief, Operator Licensing Branch
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Question 15	Key Answer / Comments	Recommendations
During performance of flow balance surveillance testing on the Normal Service Water System, several valves that are normally full open need to be throttled for the duration of the test. Which of the following describes how the change in status of these valve will be tracked? a. The throttle valve computerized listing will be updated and a Change Notice (CN) for the valve realignment will be completed for SOP-0018, "Normal Service Water". b. A Procedure Revision for the valve realignment will be completed for SOP-0018, "Normal Service Water" and an entry made in the Control Room log. c. The repositioning of the valves will be documented per the guidance in ADM-0022, "Conduct of Operations". d. The repositioning of the valves will be controlled by tracking all manipulations as required in the surveillance test package.	**Answer:** d **Reference:** ADM-0015 ADM-0022 **Comments:** Response (c) could also be correct. c. is correct for **non** Tech. Spec. testing activities (ADM-0022). d. is correct for Tech. Spec. testing activities (ADM-0015). (continued next page)	Accept both answers (c) and (d).

Question 15	Comments	Recommendations
	The use of the term "Normal Service Water" implies the manipulation of Non Safety-Related components which could be documented per ADM-0022, "Conduct of Operations." The use of the term "surveillance testing" implies the manipulation of Safety-Related components during surveillance testing which are documented per ADM-0015, "Surveillance Testing." Given the disparity in the stem of the question, ADM-0015 is no longer the ONLY reference to govern this unusual operation. ADM-0022, "Conduct of Operations" provides guidance for operational activities and therefore becomes an expected response. • ADM-0022, Section 6.3 "System Check Lists," could provide guidance, "... system checklists are used for establishing and confirming the status of equipment and systems. ... proper system configuration is assured and maintained by use of administrative controls (i.e., SOP, MDL, OSP-0014, Temp. Alt.)." • ADM-0015, Step 7.11.5 would provide guidance for repositioning valves required to be manipulated as part of the surveillance test package.	

6.2.2. Routine procedural actions that are frequently performed may not necessitate the use of a procedure. If there is any doubt as to the procedural action by the individual performing the job, the procedure must be in hand.

6.2.3. Entry conditions of emergency procedures shall be committed to memory by licensed personnel.

6.2.4. The red diagonal lines on selected annunciator windows, in the Control Room, are installed as a visual aid. These red diagonal lines denote annunciators requiring review of the EOP's for possible entry conditions.

6.2.5. If an evolution is suspended for an extended period of time, reverification of the initial conditions is required as determined by the OSS/CRS.

6.2.6. In the event that an original procedure checklist, STP or tagging order becomes contaminated the information should be transferred to a clean copy of the document. This should be noted at the top of the document along with the name of the individual who is transferring the data.

→ 6.3 System Check Lists (Ref. 2.54)

NOTE

During Refueling Outages all safety-related valve lineups shall be performed. (Ref. 2.44)

6.3.1. System checklists are used for establishing and confirming the status of equipment and systems. System checklists are performed as a prerequisite for system operation and periodically for equipment status verification.

1. Following performance of system checklists, proper system configuration is assured and maintained by use of administrative controls. (i.e., SOP, MDL, OSP-0014, Temp Alt.)

6 PLANT OPERATING PROCEDURES

6.1 Procedure Compliance

6.1.1. Plant equipment shall be operated in accordance with current written approved procedures.

6.1.2. If the individual actually performing the activity cannot or believes he should not follow the procedure governing that activity as written, he shall place the system/component into a stable and safe condition and inform the responsible supervisor. The supervisor shall resolve the discrepancy in the procedure by either:

1. Determining the methods by which the activity can be performed using the procedure as written and conveying this to the individual performing the activity, or

2. Submitting a procedure change, either temporary or permanent depending on the actual situation. No further procedural steps shall be accomplished until the procedure change is approved in accordance with RBNP-001.

6.1.3. In the event of an emergency not covered by an approved procedure, operators shall take action so as to minimize personnel injury, damage to the facility, and to protect the health and safety of the public. Such deviations to procedures shall be approved by the OSS/CRS prior to action or deviation.

6.1.4. No interlock will be bypassed or defeated from performing its function unless it is accomplished in accordance with an approved procedure or work request.

6.2 Procedure Implementation

6.2.1. Procedures that control operations where reliance on memory cannot be trusted and where operations must be performed in a specified sequence shall be followed step-by-step with the procedure in hand. Many procedures will also require signoffs, while others may need to be referred to only occasionally.

5. Include data tables within the procedure to record As-Found and As-Left Data. When it is better to locate the data tables in one location, append the consolidated data sheets as Attachment pages to the end of the surveillance procedure.

   - The acceptance criteria or tolerances in the tables should be expressed in quantitative terms (e.g. 2 turns, 100 inch- pounds, etc.) compatible with item being read.

6. Cautions and notes applicable to specific steps inserted prior to performance of the step.

7. The acceptance criteria for the Surveillance should be specified in the process parameters the test uses.

7.11.5 A restoration section should contain the following:

C → 1. Detailed instructions for restoration of the component(s) to its required configuration.

   - Include verification of the operability of the component, if practical.

C   - Independent verification shall be performed on safety-related system restoration.

2. A table to provide spaces for listing the test equipment if required. All test instruments used to record data in the test shall be listed by M&TE I.D. Number and calibration due date.

3. A channel check of the instrument for restoration verification. Normally a table of like channel comparison data.

C 4. Step for notifying the Nuclear Control Operator of test completion.

C 5. Step for notifying the OSS/CRS of test completion.

C 7.12 Acceptance Criteria

7.12.1 Acceptance Criteria is that criteria which is required by the governing document.

Question 34	Key Answer / Comments	Recommendations
Given the following conditions: - The HPCS Diesel Generator had been in a normal standby lineup - An automatic start signal has just been received - The diesel generator has responded as designed - It has been determined that NO VALID start signal condition exists What are the required IMMEDIATE actions for these conditions? a. Depress the local panel Emergency Stop pushbutton and place the Engine Mode Control Switch in "TEST". b. Depress the Main Control Room Emergency Stop pushbutton and direct the local operator to place the Engine Mode Control switch in "MAINT". c. Place the Diesel Generator Mode Switch to "LOCAL MANUAL" and direct the local operator to place the Diesel Generator Control switch to "STOP". d. Verify the Diesel Generator Mode Switch in "REMOTE MANUAL" and place the Main Control Room Diesel Generator Control switch to "STOP".	**Answer:** b **Reference:** SOP-0052, "HPCS Diesel Generator", Rev. 12, Section 2.1.27, Page 6 LO - HLO-075-3 #10.b.3 **Comments:** The question stem does not provide adequate information for the candidate to identify a single correct response. The conditions are presented in such a way that two interpretations are possible. First case, as the question was originally intended, an automatic start signal is "locked in" and has been determined to not be valid. In this condition, the only way to stop the machine is to use the emergency stop pushbutton, as identified in answer "b" (placing the mode switch in "MAINT" would prevent subsequent restart; precaution 2.1.27 of SOP-0052) (continued next page)	Throw out question since there are 3 possible correct answers.

Question 34	Key Answer / Comments	Recommendations
	Second case, although an automatic start signal has been received, the statement in the stem "It has been determined that NO VALID start signal condition exists" implies no start signal or condition is currently present. In this case, use of the emergency trip button is not appropriate (see SOP-0052 precaution 2.1.22) and not required to stop the machine. Both "c" and "d" would stop the diesel under the circumstances described. Note: The Diesel Generator mode switch positions on the panel are "REMOTE" and "LOCAL" instead of "REMOTE MANUAL" and "LOCAL MANUAL". However, since both positions simply set up the manual control functions, this level of discrimination is not appropriate for determining the distracters are incorrect. (i.e. to eliminate "c" and "d" the candidate would need to know the exact switch nomenclature).	

4. Generator reverse power.

5. Generator overcurrent

6. Engine low lube oil pressure

7. Engine high temperature (jacket cooling water)

8. Engine overcrank cycle ≥20 seconds with no start

2.1.21 Anytime the diesel engine has tripped automatically, the operator is required to depress the "STOP" pushbutton on 1E22*PNLS001 engine panel or by using the control switch on 1H13*P601, (depending on where control is). This will drop out the K1 relay which seals in upon a start. Failure to do so will result in an automatic restart of the engine when the trip condition clears.

2.1.22 Depressing the local or remote EMERGENCY STOP pushbutton will cause a diesel trip even with an emergency start signal present. This pushbutton should only be depressed under extreme emergencies. For normal operation the local or remote STOP pushbutton should be used.

2.1.23 The 1DTM*V62 TURBOCHARGER AIR BOX DRAIN VALVE must be maintained throttled open during all operations involving running of the diesel engine.

2.1.24 When the diesel is running in parallel with the grid a fault on the grid could cause a loss of the bus associated with the diesel concurrent with a trip/lockout of the diesel. To reduce the chances of this occurring minimize the time spent with the diesel paralleled to the grid.

2.1.25 Operating data pertaining to diesel generator start attempts shall be obtained per PEP-0026. DIESEL GENERATOR TRENDING AND FAILURE REPORTING.

2.1.26 During diesel fuel oil unloading, a fire watch shall be stationed at the unloading area and two (2) 150 pound dry chemical extinguishers placed near unloading area.

2.1.27 If the diesel receives an inadvertent auto start signal that is determined not to represent a valid emergency start condition, immediately depress the local or remote EMERGENCY STOP pushbutton then place the local ENGINE CONTROL switch in the MAINT position. After the signal source has been identified and the auto start circuits reset, depress the local or remote STOP pushbutton prior to placing the local ENGINE CONTROL switch in the AUTO position when returning the diesel generator to a standby mode. Valid auto start signals are indicated as follows:

1. An ECCS auto start signal is identified at 1H13*P601 by the white indicating light at the HPCS INITIATION RESET pushbutton being illuminated and at 1E22*PNLS001 by the white indicating light above the LOCKOUT RELAY being off.

4.1.15. Establish CRD System flow with isolated HCU/CRDs as follows:

NOTE

A sufficient number of HCUs must be aligned to accept cooling water flow at the minimum flow rate expected through the C11-F002A(B), CRD Flow Control Valve. Each CRD will accept about 1/3 gpm at approximately 15 psid. For example, if 10 gpm system flow is expected with C11-F002A(B) fully closed, 30 HCUs must be aligned for cooling flow to limit cooling header pressure to 15 psid. A higher flow rate or fewer HCUs will result in an undesirable higher cooling header differential pressure.

1. Align cooling water to at least 30 HCUs by opening as a minimum the valves shown on Attachment 5, Post Maintenance Unisolated CRDs, to allow for sufficient cooling water flow for CRD Pump start. The number of HCUs may be increased or decreased at OSS/CRS discretion. The following HCUs should be included in this group:

   1) East Side (270 deg) - 04-33, 04-29, 04-25, 04-21, 04-17, 08-13, 12-13, 16-09, 12-09, 24-05, 20-05, and 16-05

   2) West Side (90 deg) - 52-37, 52-33, 52-29, 52-25, 52-21, 52-17, 48-17, 48-13, 44-09, 44-13, 40-09, and 36-09

CAUTION

To prevent possible damage to an unvented CRD from an inadvertant scram, any HCU or CRD which has been opened for maintenance should remain isolated from the charging header with HCU-XXXX-V113 CHARGING WATER RISER ISOL VLV closed and HCU-XXXX-V107, SCRAM ACCUMULATOR DRAIN VLV opened one turn. Do not unisolate charging water to the CRD until it has been vented.

2. Verify charging water is isolated to any accumulators which were opened for maintenance and document this verification on Attachment 6, Post Maintenance Isolated Accumulators.

3. IF the Reactor is depressurized, THEN open C11-VF062, DR WTR PC STATION RELIEF LINE TO R.B. DRNS ISOL VLV.

4. Start C11-C001A(C001B), CRD PUMP A(B).

CAUTION

To prevent possible damage to an unvented CRD from an inadvertant scram, any HCU or CRD which has been opened for maintenance should remain isolated from the charging header with HCU-XXXX-V113, CHARGING WATER RISER ISOL VLV closed, and HCU-XXXX-V107, SCRAM ACCUMULATOR DRAIN VLV opened one turn. Do not unisolate charging water to the CRD until it has been vented.

4.1.13. Verify charging water is isolated to any accumulators which were opened for maintenance and document this verification on Attachment 6, Post Maintenance Isolated Accumulators.

4.1.14. Establish normal CRD System flow as follows:

1. Start C11-C001A(C001B), CRD PUMP A(B).

2. Slowly open C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV.

NOTE

It may take several minutes for the accumulators to charge.

3. Check the accumulators are charged by observing:

- At H13-P680, WHEN the Accumulator Fault Pushbutton is depressed, THEN All ACCUMULATOR lights on the full core display are off for the in service accumulators.

- IF all accumulators are in service, THEN ACCUMULATOR TROUBLE annunciator at H13-P680 is clear.

4. Using C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B in MANUAL mode, and C11-F003, CRD DRIVE WATER PRESS CONTROL VALVE, establish a Drive Water Differential Pressure of 250 psid at a flow rate of 41 to 49 gpm.

5. Go To Step 4.1.16.

4.1.3. Check the oil level for CRD Pump A(B) and its associated gear box is normal.

4.1.4. Vent the online CRD Pump Suction Filter using DER-V3002(V3004), C11-FLT010A(B) VENT VALVE.

4.1.5. Vent both the CRD Pump A and B casings using C11-VF109A and B, CRD PUMP A and B CASING VENT VLV.

4.1.6. Vent the online CRD Pump Discharge filter using C11-VF022A(B), CRD PMP DISCH FILTER D003A(B) VENT VLV and DER-V8(V10), C11-FLTD003A(B) VENT VALVE.

4.1.7. Start C11-C001AP(BP), CRD AUX OIL PUMP A(B).

4.1.8. Verify C11-C001A(B), CRD PUMP A(B) white light comes on.

4.1.9. Close, then throttle open C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV open one turn off the closed seat.

4.1.10. Place C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B in MANUAL at 0% output.

4.1.11. Verify suction pressure is at least 15 psig on local indicator C11-R017A(B).

NOTE

All 145 HCUs are not required to be in a normal valve lineup prior to starting the CRD pumps. HCUs which have been isolated for maintenance cannot be fully returned to service until after the CRD pump is operating.

4.1.12. IF more than 12 HCU/CRDs were isolated for maintenance, THEN Go To Step 4.1.15 for HCU alignment and CRD Pump start.

3.2 Check Condensate Storage, Makeup, and Transfer is in operation per SOP-0008, Condensate Storage, Makeup, and Transfer System.

3.3 Check Drywell and Containment Leak Detection is in operation per SOP-0033, Leak Detection.

3.4 Check Reactor Plant Component Cooling Water is in operation per SOP-0016, Reactor Plant Component Cooling Water System.

3.5 Check Nuclear Boiler Instrumentation is in operation per SOP-0001, Nuclear Boiler Instrumentation.

3.6 Check Reactor Protection is in operation per SOP-0079, Reactor Protection System.

3.7 Check the following electrical systems are in operation:

3.7.1. 4160 VAC per SOP-0046

3.7.2. 480 VAC per SOP-0047

3.7.3. 120 VAC per SOP-0048

3.7.4. 125 VDC per SOP-0049

3.8 Verify B33-MOVF023A and B, Recirc Pump Suction Valves are open to prevent overpressurization of the Recirculation Pump Piping.

NOTE

All 145 HCUs are not required to be in a normal valve lineup prior to starting the CRD pumps. HCUs which have been isolated for maintenance cannot be fully returned to service until after the CRD pump is operating.

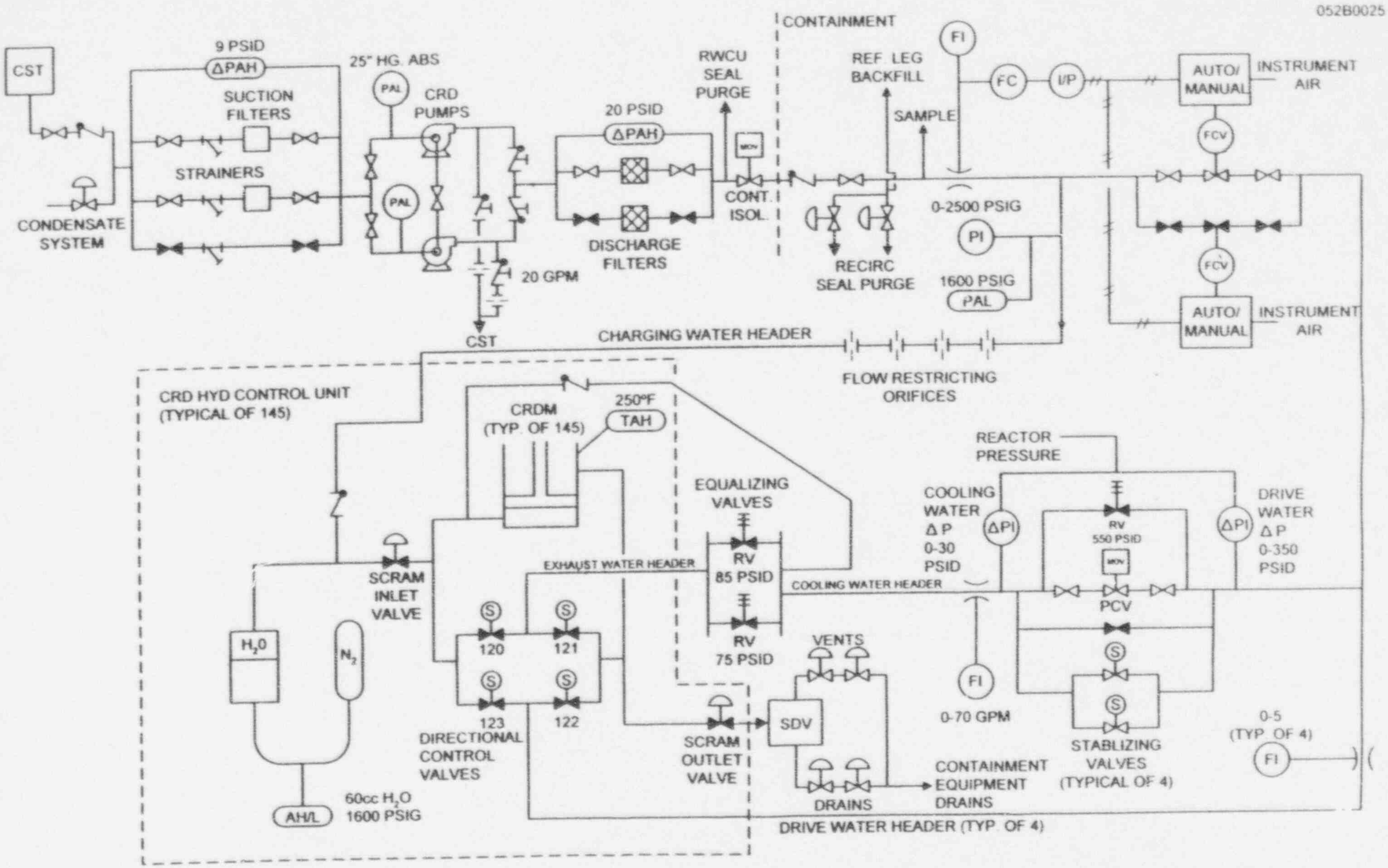
3.9 Verify system is lined up for startup.

4 SYSTEM STARTUP

4.1 Placing the CRD System in Operation

4.1.1. Charge all HCU Accumulators which are lined up for startup with nitrogen per Section 4.3.

4.1.2. Verify that the Reference Leg Backfill System has been isolated per SOP-0001.



CONTROL ROD DRIVE HYDRAULIC SYSTEM

RBS-1-STM-GPST-A0052.00

FIGURE1

3. Remote Controls and Indications

- Remote Flow Indicator C11-FIR606 (H13-PNLP601) - 0-100 GPM
- Flow Controller C11-FCR600 (H13-PNLP601) Manual/Auto control
- Flow Controller Valve C11-FVF002A/B (H13-PNLP601) Indicating Lights-GREEN-CLOSED, RED-OPEN

4. Automatic Features

Valve fails closed on loss of instrument air

1.3.2.7 Drive Water Header (Figure 1)

1. Description

There are 4 drive water headers each is equipped with a stabilizing unit. The stabilizing valves are controlled automatically by RC&IS. They will close during control rod movement to divert water from the cooling water header to the drive water header. This is done to maintain a constant system pressure during rod movement. The stabilizing valve units consist of two normally open parallel solenoid operated valves. Needle valves down stream of each stabilizing valve are set such that each valve passes 2 gpm. On a control rod insert both stabilizing valves close which provides a 4 gpm flow to the bottom of the drive piston. On a control rod withdraw one stabilizing valve closes which provides a 2 gpm flow to the top of the drive piston.

→ Drive water pressure is manually controlled by Pressure Control Valve C11-MOVF003 at reactor pressure plus 250 psig. Drive water pressure may also be controlled by a manual bypass around the Pressure Control Valve. Should drive water exceed 550 psid a relief valve opens relieving to the cooling water header. This feature avoids the possibility of excess water pressure causing excessive control rod drive speeds.

Question 41	Comments	Recommendations
	SOP-0002, "Control Rod Drive System" 4.1 Section "Placing the CRD System in Operation," states that system flow and drive pressure will need to be adjusted to obtain a d/p of 250 psid. Throughout this procedure, during normal system operations and evolutions, drive pressure will need to be adjusted to obtain and maintain a d/p of 250 psid. STM-A0052.00 states that, "Drive water pressure is manually controlled by PCV at reactor pressure plus 250 psig."	

<table>
<tr><th>Question 41</th><th>Key Answer / Comments</th><th>Recommendations</th></tr>
<tr><td>Prior to a startup with the plant in a cold, depressurized condition, the Unit Operator adjusts the Control Rod Drive Hydraulic (CRD) system Pressure Control Valve (PCV) to maintain 250 psid between drive water header pressure and reactor pressure.

How is this pressure differential maintained as reactor pressure increases during the ensuing startup to 100% power? (Assume the CRD system is operating as designed.)

As reactor pressure increases during the startup:

a. the Pressure Control Valve automatically operates to maintain CRD system pressure above reactor pressure.

b. the Unit Operator must periodically open the Pressure Control Valve to maintain the required differential pressure.

c. the Flow Control Valve automatically opens to maintain a constant flow, therefore a constant d/p across the PCV.

d. the Unit Operator must periodically adjust the Flow Control Valve to maintain CRD system pressure above reactor pressure.</td><td>Answer: c

Reference:

STM-A0052.00
SOP-0002

Comments:

No correct answer.

Answer (c) assumes that the system is operating differently than the installed design. If the CRD system is operating as installed (the design has not been changed), then the Pressure Control Valve must be periodically throttled closed to maintain the differential pressure between drive water header pressure and reactor pressure. During a plant startup, the Pressure Control Valve <u>must be throttled in the closed direction</u> to maintain the differential pressure between CRD water and reactor pressure, as reactor pressure increases.

(continued next page)</td><td>Throw out this question, since there is no correct answer.</td></tr>
</table>

2. An undervoltage auto start signal is identified at 1H13*P601 by the DIV III 4KV BUS UNDERVOLTAGE annunciator being in alarm. There is no indication of an undervoltage auto start signal at 1E22*PNLS001.

2.2 Engine Cooling System

2.2.1 Do not open expansion tank filler cap unless the system is cool. Then partially open the filler cap to relieve pressure prior to complete removal of the cap.

2.2.2 Ensure Immersion Heater control switch is in "OFF" anytime the cooling system or Immersion Heater is not filled with coolant.

2.3 Fuel Oil System

2.3.1 Anytime work is done on the fuel oil day tank level instrumentation the control switch for the Fuel Oil Transfer Pump must be placed in "OFF" to preclude pumping diesel fuel to the roof.

2.3.2 If the diesel is run for one hour or longer, check and drain from the day tank any accumulated water via valve 1EGF*V71 DAY TANK DRAIN.

2.4 Lube Oil System

2.4.1 Do not remove round caps from lube oil strainer housing with the engine running.

2.4.2 Before attempting to start a new engine or one that has had internal maintenance performed a prelubrication must be performed in accordance with the EMD Vendor's Manual.

2.4.3 Before any Normal start of the standby diesel proper prelubrication shall be verified in accordance with Section 4.2.1.

2.4.4 The engine oil sump level should be between the LOW and FULL marks on the dipstick with the engine running. The level should be just above the 3/4 FULL mark with the engine idle.

2.4.5 The lube oil system level should be between the upper and lower sightglasses with the engine shutdown and above the upper sightglass with the engine running.

2.4.6 After changing oil filter elements or performing any evolution which involves draining the accessory oil system of the diesel, the oil system must be refilled, the strainer box verified full, and the SOAKBACK (Turbo) LUBE OIL pumps operated for at least 30 minutes prior to starting the engine. The engine should then be brought to an unloaded condition to assure complete filling of accessories before any subsequent loading is done.

2.4.7 Oil level in the engine governor should be maintained at or above the fill mark on the sightglass with the diesel shutdown. With diesel in operation the suggested level is $\pm$ 1/4" of fill mark.

5. Close, then slowly open C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV.

NOTE

It may take several minutes for the accumulators to charge.

6. With exception of the accumulators listed on Attachment 6, Post Maintenance Isolated Accumulators; at H13-P680, WHEN the Accumulator Fault Pushbutton is depressed, THEN All ACCUMULATOR lights on the full core display are off for the in service accumulators.

7. WHEN flow has stabilized, IF C11-VF062, DR WTR PC STATION RELIEF LINE TO R.B. DRNS ISOL VLV was opened in Step 4.1.15.3, THEN close the valve while monitoring cooling header differential pressure. IF cooling header differential pressure exceeds 30 psid, THEN secure the CRD Pump. Pump startup may be re-attempted after additional HCUs are aligned for cooling flow.

NOTE

A lower flow rate may be established as required by the Operations Shift Superintendent as long as adequate cooling is provided for the CRDs.

8. Adjust system flow and drive pressure using C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B and C11-F003, CRD DRIVE WATER PRESS CONTROL VALVE to obtain a flow equal to 0.31 times the number of CRDs aligned for cooling flow and a differential pressure of 250 psid.

4.1.16. IF the Reactor is depressurized, THEN fill and vent the system in accordance with Section 4.2.

4.2 System Fill and Vent

NOTE

Fill and vent of the CRD System and HCU/CRDs is not required if the Reactor is pressurized.

4.2.1. Verify the CRD System is in operation per Section 4.1.

5 <u>SYSTEM NORMAL OPERATION</u>

5.1 Alternating CRD Pumps

<u>NOTE</u>

To minimize the potential impact of pressure transients in the CRD Hydraulic System, shifting CRD Pumps when Reactor pressure is greater than 0 psig but less than full operating pressure should be avoided except in extraordinary circumstances.

5.1.1. Perform the following steps for the CRD Pump to be started.

1. Check that the oil level for CRD Pump A(B) and its associated gear box is normal.
2. C Close C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV.
3. Vent the CRD Pump A(B) casing using C11-VF109A(B), CRD PUMP A(B) CASING VENT VLV.
4. Start C11-C001AP(BP), CRD AUX OIL PUMP A(B).
5. Verify white light comes on for C11-C001A(B), CRD PUMP A(B).
6. C Start C11-C001A(B), CRD PUMP A(B) and immediately, but slowly, open C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV.
7. Verify CRD Pump A(B) is operating properly.

5.1.2. Check that CRD System flow has stabilized.

5.1.3. Perform the following steps for the CRD Pump to be stopped:

1. Close C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV.
2. Stop C11-C001A(B), CRD PUMP A(B).

5.1.4. Verify that the pressure and flow of the CRD system are normal. <u>IF</u> they are <u>NOT</u>, <u>THEN</u> use C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B and C11-F003, CRD DRIVE WATER PRESS CONTROL VALVE to obtain a system flow rate as needed to establish an indicated value for Drive Water Differential Pressure of 250 psid and a flow rate of 41 to 49 gpm.

5.1.5. Slowly reopen C11-VF014A(B), CRD PMP A(B) DISCH STOP CHECK VLV for the CRD Pump which was stopped.

5.2 Alternating Flow Control Valves

NOTE

Changeover of the Flow Control Valves is performed at the Local Flow Control Stations, C11-D009A(B) unless otherwise noted.

5.2.1. Verify C11-VF127A(B), INST AIR SPLY TO C11-FV-F002A(B) is open.

5.2.2. Verify the following for the flow control valve to be placed in service:

1. Flow Control Valve AUTO/MANUAL Transfer Switch in MANUAL.
2. Manual control knob adjusted to its minimum position.

5.2.3. Open C11-VF046A(B), FLOW CONT VLV F002A(B) INLET ISOL VLV, for the flow control valve to be placed in service.

5.2.4. Open C11-VF047A(B), FLOW CONT VLV F002A(B) OUTLET ISOL VLV, for the flow control valve to be placed in service.

5.2.5. Slowly open the standby C11-FV-F002A(B), CRD FLOW CONTROL VALVE, and observe the previously inservice C11-FV-F002A(B), CRD FLOW CONTROL VALVE, closes.

5.2.6. WHEN the previously inservice flow control valve is fully closed, THEN adjust its manual control knob to minimum position, and place its AUTO/MANUAL Transfer Switch in MANUAL.

5.2.7. Verify that the automatic and manual signals of the flow control valve being placed in service are matched. Then place the AUTO/MANUAL Transfer Switch in AUTO.

5.2.8. Close C11-VF047A(B), FLOW CONT VLV F002A(B) OUTLET ISOL VLV, for the flow control valve being removed from service.

5.2.9. Close C11-VF046A(B), FLOW CONT VLV F002A(B) INLET ISOL VLV, for the flow control valve being removed from service.

5.2.10. Notify Control Room that changeover is complete.

→ 5.2.11. In the control room, verify that the pressure and flow of the CRD system are normal. IF they are not, THEN use C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B and C11-F003, CRD DRIVE WATER PRESS CONTROL VALVE to obtain a system flow rate as needed to establish an indicated value for Drive Water Differential Pressure of 250 psid and a flow rate of 41 to 49 gpm.

5.3 Alternating Discharge Filters

5.3.1. Verify the C11-PDIS-N002, Filter Differential Pressure Indicating Switch is in operation.

5.3.2. Perform the following for the filter to be placed in operation.

1. Open C11-VF020A(B), CRD PMP DISCH FILTERS D003A(B) INLET ISOL VALVE.
2. Open C11-VF022A(B), CRD PMP DISCH FILTER D003A(B) VENT VLV and DER-V8(V10), C11-FLTD003A(B) VENT VALVE. WHEN air-free water is vented from filter, THEN close C11-VF022A(B), VENT VALVE and DER-V8(V10), C11-FLTD003A(B) VENT VALVE.
3. Slowly open C11-VF021A(B), CRD PMP DISCH FILTER D003A(B) OUTLET ISOL VLV.

5.3.3. Perform the following for the filter being removed from service.

CAUTION

Backflow through the filter could damage the filter cartridge. Do not close C11-VF020A(B) before closing C11-VF021A(B).

1. Close C11-VF021A(B), CRD PMP DISCH FILTER D003A(B) OUTLET ISOL VALVE.
2. Verify proper filter operation by observing C11-PDIS-N002 is in normal band.
3. Close C11-VF020A(B), CRD PMP DISCH FILTER D003A(B) INLET ISOL VALVE.

5.3.4. Drain the filter removed from service by performing the following:

1. Open the following valves:

   1) C11-F022A(B), CRD PMP DISCH FILTER D003A(B) VENT VLV

   2) C11-VF023A(B), CRD PMP DISCH FILTER D003A(B) DRAIN VLV

   3) DER-V8(V10), C11-FLTD003A(B) VENT VALVE

   4) DER-V9(V11), C11-FLTD003A(B) DRAIN VALVE

2. <u>WHEN</u> filter is completely drained, <u>THEN</u> close the following valves:

   1) C11-F022A(B), CRD PMP DISCH FILTER D003A(B) VENT VLV

   2) C11-VF023A(B), CRD PMP DISCH FILTER D003A(B) DRAIN VLV

   3) DER-V8(V10), C11-FLTD003A(B) VENT VALVE

   4) DER-V9(V11), C11-FLTD003A(B) DRAIN VALVE

5.3.5. Notify Control Room that changeover is complete.

→ 5.3.6. In the control room, verify that the pressure and flow of the CRD system are normal. <u>IF</u> they are <u>NOT</u> , <u>THEN</u> use C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B and C11-F003, CRD DRIVE WATER PRESS CONTROL VALVE to obtain a system flow rate as needed to establish an indicated value for Drive Water Differential Pressure of 250 psid and a flow rate of 41 to 49 gpm.

5.4 Alternating Suction Filters

5.4.1. Verify that the C11-PDIS-N015, Filter Differential Pressure Indicating Switch is in operation.

5.4.2. Perform the following for the filter to be placed in operation.

1. Open C11-VF115A(B), CRD PMP SUCT FILTER D010A(B) INLET ISOL VLV.
2. Slowly open DER-V3002(V3004), C11-FLT010A(B) VENT VALVE.
3. When air-free water is vented from the filter, close DER-V3002 (V3004), C11-FLT010A(B) VENT VALVE.
4. Slowly open the C11-VF114A(B), CRD PMP SUCT FILTER D010A(B) OUTLET ISOL VLV.

5.4.3. Perform the following for the filter being removed from service.

CAUTION
Backflow through the filter could damage the filter cartridge. Do not close C11-VF115A(B) before closing C11-VF114A(B).

1. Close C11-VF114A(B), CRD PMP SUCT FILTER D010A(B) OUTLET ISOL VLV.
2. Verify proper filter operation by observing C11-PDIS-N015 is in normal band.
3. Close C11-VF115A(B), CRD PUMP SUCT FILTER D010A(B) INLET ISOL VLV.

5.4.4. Notify Control Room that changeover is complete.

→ 5.4.5. In the control room, verify that the pressure and flow of the CRD system are normal. IF they are not, THEN use C11-R600, CRD HYDRAULICS FLOW FLOW CONTROLLER C11-F002A/B and C11-F003, CRD DRIVE WATER PRESS CONTROL VALVE to obtain a system flow rate as needed to establish an indicated value for Drive Water Differential Pressure of 250 psid and a flow rate of 41 to 49 gpm.

Question 46	Key Answer / Comments	Recommendations
Which of the following is indication that a failure of both seals on one recirculation pump have occurred? a. Drywell identified leakage will decrease and unidentified leakage will increase. b. Drywell identified leakage will increase and unidentified leakage will increase. c. Drywell identified leakage will increase and unidentified leakage will decrease. d. Drywell identified leakage will decrease and unidentified leakage will decrease.	**Answer:** b **Reference:** STM-A0053.01 **Comments:** Response (a) could also be correct. The amount of leakage (where and how detected) depends on the location and severity of the seal failures. Response (a) would be correct for a catastrophic failure of the outer seal with a minor failure of the inner seal. In this case, flow to the Drywell Equipment Drain Sump (identified leakage) is reduced due to a drop in pressure between the seals caused by less resistance to flow into the Drywell atmosphere (unidentified leakage). Response (b) would be correct for a minor failure of the outer seal. This is due to an increase in pressure between the seals where flow to the Drywell Equipment Drain Sump (identified leakage) would increase with a corresponding increase in leakage to the Drywell atmosphere (unidentified leakage) due to the minor failure of the outer seal.	Accept both answers (a) and (b).

1.3.5.3 Remote Controls and Indications

The indicators on H13-PNLP680 show the pressure drop across each seal during normal operation (see Figure 5). During normal operation, No. 1 seal cavity pressure should read approximately 10-15 psig greater than RPV pressure, and No. 2 Seal Cavity pressure should read approximately 1/2 the value of seal cavity No. 1. If the No. 1 seal has a gross failure, seal cavity No. 2 pressure increases to the value of seal cavity No. 1 and RECIRC PUMP A(B) SEAL STAGING HIGH/LOW FLOW (H13-PNLP680) alarms. Should seal No. 2 fail, the RECIRC PUMP A (B) OUTER SEAL HIGH LEAKAGE annunciator (H13-PNLP680) alarms, the No. 2 Seal Cavity pressure lowers, and the RECIRC PUMP A(B) SEAL STAGE HIGH/LOW FLOW (H13-PNLP680) alarms.

If both seals fail, the pressure in cavity #1 remains about the same. The pressure of cavity #2 may go higher or lower depending on which seal is worse. Additionally, at this point, there is significant seal leakage, which is indicated by a rise in unidentified leakage. Both the RECIRC PUMP A(B) SEAL STAGING HIGH/LOW FLOW annunciator (H13-PNLP680) and the RECIRC PUMP A(B) OUTER SEAL HIGH LEAKAGE annunciator (H13-PNLP680) alarm.

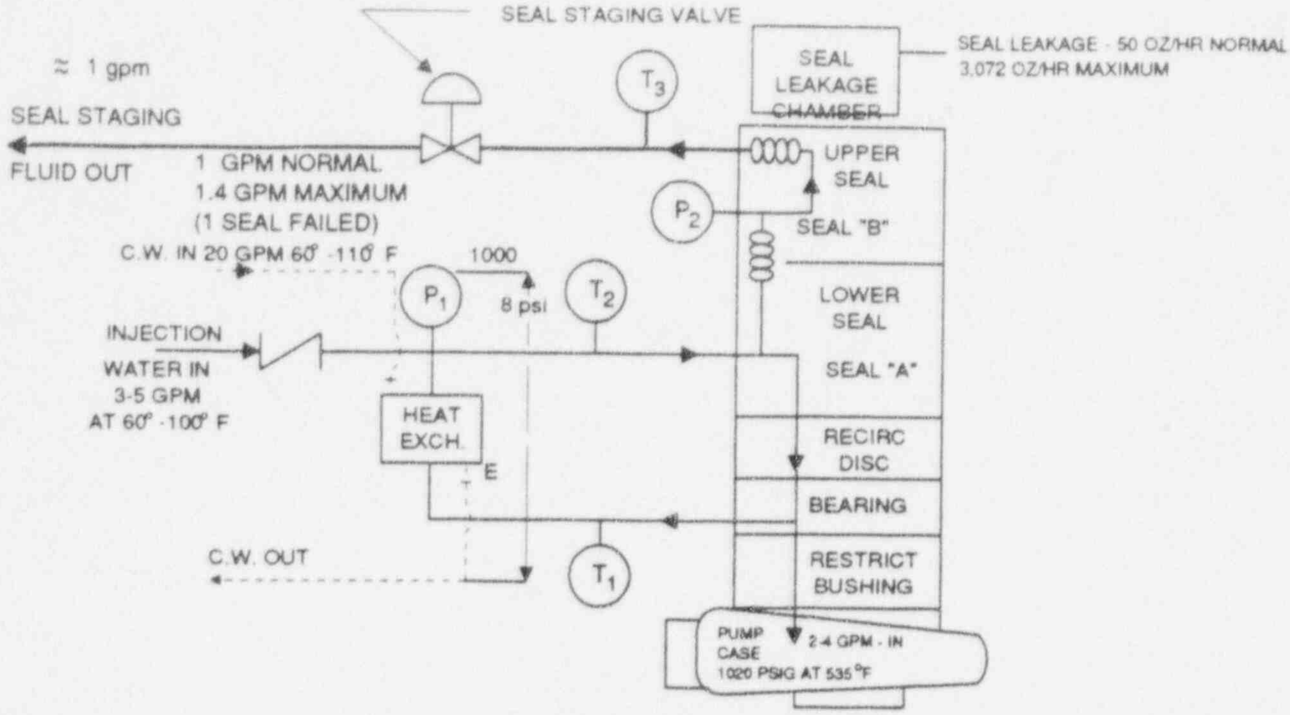
It is also possible to have a partial seal failure which would cause pressure indications to respond as described above but not cause an alarm actuation.

1.3.5.4 Automatic Functions

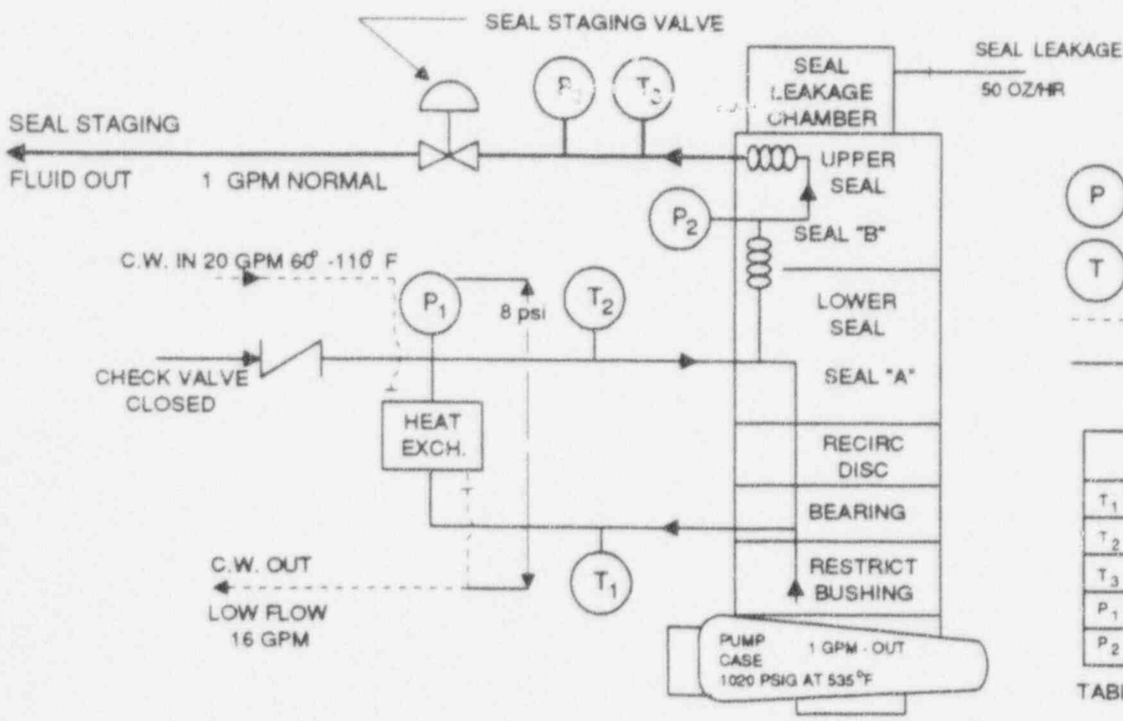
The Reactor Recirculation Pump A(B) Seal Flow Outlet Isolation Valve (B33-FVF075A(B)) is now operable. The valves are OPEN regardless of switch position if the respective recirc pump is on line. The valves open if OPEN switch is depressed and the respective pump is off-line. The valves closes when AUTO switch is depressed and the respective pump is off-line.

1.3.5.5 Power Supplies

N/A



CONDITION A - NORMAL



LEGEND

P PRESSURE TAP

T THERMO COUPLE

- - - - COOLING WATER

——— REACTOR WATER

	WITH INJECTION	WITHOUT INJECTION
T_1	115 °F	150 °F
T_2	112 °F	138 °F
T_3	127 °F	153 °F
P_1	1020 PSIG	1020 PSIG
P_2	510 PSIG	510 PSIG

TABLE BASED ON
INJECTION TEMP = 110° F
COOLING WATER TEMP = 105° F

CONDITION B - LOSS OF INJECTION

RECIRCULATION PUMP SEAL FLOWS

the hotwell increases cooling load on RWCU non–regenerative heat exchangers. That increased load will increase the cooling water temperature to the recirc pump seals. This has caused pressure oscillation of up to 70 psig on second stage seals.

7.3.3 Drywell

- Failure of seals or excessive pump seal leakage will result in increased drywell pressure and increased load on the drywell coolers.
- Excessive pump seal leakage increases unidentified leakage in the drywell.

7.3.4 Venting

- Inadequate venting of the seals has been a significant problem affecting seal life.

TP–06

7.3.5 Normal seal cavity pressure indications will show No. 1 seal pressure approximately 10–15 psig greater than reactor pressure.

- No. 2 seal pressure will indicate approximately 1/2 the value of No. 1 seal pressure.

OBJ #10

7.4 Technical Specifications applicable to the recirculation system include:

7.4.1 ATWS Recirculation Pump Trip System Instrumentation (3/4.3.4.1).

7.4.2 EOC–RPT System Instrumentation (3/4.3.4.2).

7.4.3 Recirculation Loop (3.4.1.1).

7.4.4 Jet Pumps (3.4.1.2)

7.4.5 Recirculation Loop Flow (3.4.1.3)

7.4.6 Idle Recirculation Loop Startup (3.4.1.4)

OBJ #14

8.0 Pump seal failure

8.1 Inner seal failure indications:

TP–06

- This will be indicated by seal No. 2 pressure rising to equal seal No. 1 pressure.

Question 59	Key Answer / Comments	Recommendations
Following a normal reactor scram, at what point must the At-The-Controls Operator be alert for narrow range reactor water level indication "outgassing" or "notching"? (Assume the Reference Leg Backfill system is NOT in service.) a. If the cooldown rates exceed the Technical Specification limits. b. If the MSIVs close and pressure control is on the Safety Relief Valves. c. When the last Feed Pump is removed from service. d. As reactor pressure decreases below 450 psig.	**Answer:** d **Reference:** GOP-0002 **Comments:** Response (a) and (b) are also correct. GOP-0002, P&L 1.29, "If the Reference Backfill System is not in service, then notching ... may occur as reactor pressure is lowered. A slow cooldown rate will decrease the effect of degassing the reference leg." a. To exceed Tech. Spec. limit of 100°F/hr, pressure would have to decrease rapidly from 1000psig to below 400psig, and therefore outgassing could occor. b. Opening SRVs without EHC system compensation (MSIVs closed) is also rapid depressurization (a condition that also promotes outgassing). d. industry experience indicates that notching may occur ~ 450 psig and the ATC must be alert.	Throw out question since there are 3 possible correct answers.

1.27 During Hot Standby conditions when feedwater flow rate is low, operation of the RWCU system may cause feedwater line thermal stratification. Feedwater lines may be monitored for thermal stratification per SOP-0110 TAMARIS TEMPERATURE SCANNER Display 33. If feedwater line thermal stratification is indicated by a delta temperature between the top and bottom of the feedwater line of greater than or equal to 100°F refer to SOP-0009 REACTOR FEEDWATER SYSTEM.

1.28 Minimize the time between reactor recirculation pump transfer to slow and stopping the second reactor feed pump.

1.29 If the Reference Leg Backfill System is not in service, then notching of reactor level indication may occur as reactor pressure is lowered. A slow cooldown rate will decrease the effect of degassing the reference leg.

1.30 If the Reference Leg Backfill System is not in service, then a reactor level bias could develop from degassing of the level reference leg large enough to effect reactor level trips/isolations.

1.31 Reactivity Controls During Reactor Shutdowns

1. All control rod insertions shall be done using the Shutdown Control Rod Sequence Package unless otherwise recommended by the on-duty reactor engineer.

2. Reactor power should be reduced with core flow prior to control rod insertions to minimize fuel duty.

3. If substitute values are entered into the process computer that affect thermal limits, the on-duty reactor engineer should update these values as necessary during power reductions to ensure accurate thermal limit monitoring.

4. Prior to reducing power below the low power setpoint, the on-duty reactor engineering should verify that all control rods are properly aligned per the BPWS constraints.

5. The reactor should be scrammed from a power level at which normal pressure control is available (i.e., turbine control valves open) to preclude inadvertent or uncontrolled reactivity changes due to reactor cooldown and depressurization.

6. A reactor shutdown by manual control rod insertion of all control rods shall not be permitted unless specifically authorized by the Operations Superintendent.

STEP		INITIALS
1.	Notify System Operator prior to decreasing generator load.	
2.	If the Reference Leg Backfill System is not in service per SOP-0001, then have I&C stage equipment, acquire necessary technicians and obtain PMs to backfill reactor water level reference legs. (Approximately 12 hours may be needed to prepare for backfilling.) Actual backfilling performance may commence when Operations Shift Superintendent authorizes. (It is desired to have backfilling completed prior to reactor pressure reaching 450 psig to counter level indication notching possibilities.)	
3.	Monitor turbine vibration, bearing temperature and differential expansion. <u>Turbine Temp & Expansion RCDR</u> [1TMI-NXR102] a. Differential Expansion Rotor Long (point 11) between 0.31 inches and 0.69 inches. (Refer to ARP-870-54, G08, H08) b. Rotor Expansion Rotor Long (point 12) between 0.455 inches and 1.545 inches. <u>Turbine Vibration RCDR [TMI-NXR103]</u> a. Vibration (points 1 thru 10) between 0 mils and 6 mils. (Refer to ARP-870-54, D08) <u>Tamaris Computer</u> (Display 69, 70) a. Bearing oil temperatures (< setpoint, 180°F) b. Bearing metal temperature (< setpoint, 218.7°F) If unusual indications are observed, initiate hold in power change until those indications return to normal.	
4.	**NOTE** *If control rod movement is required consult the Reactor Engineer before movement.* Decrease reactor and turbine generator power by decreasing reactor recirculation flow and/or moving controls rods.	

1.0 PURPOSE

1.1 To provide instruction to the Operations Section for positioning Root Valves in the Nuclear Boiler Information System.

1.2 To provide input to the Operations Section from the Control and Instrumentation Department that the Control and Instrumentation Valves associated with the Nuclear Boiler Instruments are aligned so as to place those instruments in service.

1.3 To provide direction for operating the Reference Leg Backfill System. The function of the Reference Leg Backfill System is to purge out or keep out dissolved non-condensible gasses from the reference legs; such that, upon a depressurization non-condensible gas induced level errors will not occur. This system resolves the issues described in Generic Letter 92-04 and NRC Bulletin 93-03 concerning level errors created by non-condensible gasses coming out of solution and causing reference leg water inventory losses.

2.0 PRECAUTIONS AND LIMITATIONS

2.1 The limitations of this system are included in Technical Specifications Sections: 2.1.3, 2.1.4, 2.2.1, 3/4.3.1, 3/4.3.2, 3/4.3.3, 3/4.3.4, 3/4.3.5, 3/4.3.7.4, 3/4.3.7.5, 3/4.3.7.6, 3/4.6.1.6 and 3/4.9.8.

2.2 Relationship of the Reference Leg Backfill System with Tech Specs:

2.2.1 The Reference Leg Backfill System is considered a "support system" for the RPV Water Level Instruments.

2.2.2 The Reference Leg Backfill System is intended to be used in Operating Modes 1 through 4; however, once the reactor is depressurized the backfill system is no longer required to operate.

2.2.3 If one channel becomes inoperative or out of service then it should be returned to service within 30 days. If not returned to service within this time frame then Technical Specifications (3/4.3.1, 3/4.3.2, 3/4.3.3, 3/4.3.4, 3/4.3.5) should be consulted for required actions for an inoperable reactor water level instrument channel; and a special report, justifying continued operation, should be submitted to the NRC by the end of the 30 days.

1. Compensatory measures as discussed in NRC Bulletin 93-03 may also be taken (i.e. existing methods of backfilling/purging the reference legs: MCP-4188, 4189, 4190, 4191).

Question 80	Key Answer / Comments	Recommendations
Given the following conditions: - A plant startup is in progress with the Reactor Mode Switch in "Startup/Hot Standby" - The Outboard Main Steam Isolation Valves have just been opened and steam line warming is in progress - Main condenser vacuum has been established - A loss of the "A" RPS Bus has just occurred How will this bus loss affect the plant assuming it has NOT been restored as directed by AOP-0010, "Loss Of One RPS Bus"? a. The Outboard Main Steam Isolation Valves will begin to drift close. b. Main condenser vacuum will begin to decrease. c. The Recirculation Pumps will automatically trip. d. The Scram Discharge Volume will begin filling.	**Answer:** b **Reference:** AOP-0010 **Comments:** Response (d) is also correct. AOP-0010, "Loss Of One RPS Bus," Secetion 1.1, indicates that IAS*MOV106, Cntmt Air Isol. Valve isolates on a loss of RPS A. When IAS*MOV106 closes this isolates Instrument Air to the Containment causing containment instrument air pressure to decrease eventually bleeding off and causing the scram valves to drift open. This will fill the scram discharge volume.	Accept both answers (b) and (d).

3 AUTOMATIC ACTIONS

3.1 Loss of RPS Bus A

3.1.1. Half scram on Channel A.

3.1.2. Division 1 NSSSS isolation.

3.1.3. Standby Gas Treatment A and Annulus Mixing A start.

3.1.4. Fuel Building Charcoal Ventilation Treatment A starts.

3.1.5. Control Building Charcoal Ventilation Treatment Train A starts.

3.1.6. Control Building HVAC A re-aligns to the Charcoal Ventilation Treatment Train.

3.1.7. CMS H_2 Analyzer A auto starts.

3.1.8. ARC-P1A and ARC-P1B, COND AIR REMOVAL PUMPS trip.

3.2 Loss of RPS Bus B

3.2.1. Half scram on Channel B.

3.2.2. Division II NSSSS isolation.

3.2.3. Standby Gas Treatment B and Annulus Mixing B start.

3.2.4. Fuel Building Charcoal Ventilation Treatment B starts.

3.2.5. Control Building Charcoal Ventilation Treatment Train B starts.

3.2.6. Control Building HVAC B re-aligns to the Charcoal Ventilation Treatment Train.

3.2.7. CMS H_2 Analyzer B auto starts.

SUBSEQUENT OPERATOR ACTIONS FOR LOSS OF RPS BUS A

NOTE

Reopening of IAS-MOV106 must be done in a timely manner to prevent the Inboard MSIVs from closing

1 On H13-P601, depress the following to reset the isolation:

_____ (Initials)

- B21H-S33, INBD ISOLATION SEAL-IN RESET Pushbutton
- B21H-S32, OUTBD ISOLATION SEAL-IN RESET Pushbutton

2 At H13-P870, reopen the following isolation valves:

	RPS A	ISOLATED	RESTORED
2.1	IAS-MOV106, INST AIR OUTBD ISOL	(Initials)	(Initials)
2.2	CCP-MOV138, CONTMT SPLY OUTBD ISOL	(Initials)	(Initials)
2.3	CCP-MOV142, RR PUMP CLG SUPPLY	(Initials)	(Initials)
2.4	CCP-MOV143, RR PUMP CLG DN STREAM RTN	(Initials)	(Initials)
2.5	CCP-MOV159, CONTMT RTN OUTBD ISOL	(Initials)	(Initials)

3 IF CCP is lost to the Reactor Recirc Pumps, THEN perform the following:

3.1 IF there is a significant rise in Reactor Recirc Pump bearing or motor stator temperatures, THEN manually trip the Reactor Recirc Pumps.

_____ (Initials)

Question 89	Key Answer / Comments	Recommendations
Given the following conditions: - A reactor and plant startup is in progress with power at 2% - RPV pressure is 800 psig - The Reactor Mode Switch is in "Startup/Hot Standby" - The "A" CRD Pump has just tripped - The "B" CRD Pump is being lined up for a start - Scram accumulator 32-41 trouble alarm is in and has been verified to be from low pressure (indicating 1500 psig) Which of the following will the Control Room Supervisor (CRS) use to make the determination on declaring control rod 32-41 "Slow" or "Inoperable"? (Note: No other accumulator trouble alarms are in.) The determination is based upon: a. the control rods' last scram time surveillance results. b. the time charging water header pressure will be less than 1520 psig. c. the number of control rods fully withdrawn. d. drive water pressure availability to fully insert the control rod.	**Answer:** a **Reference:** Tech Spec 3.1.5 **Comments:** This question is too detailed for a "from memory" knowledge item. It is requiring the candidate to know, from memory, a NOTE within the Technical Specifications, as compared to a required action item associated with a Limiting Condition for Operation (LCO) with an action less than 1 hour. Typically Notes within the Technical Specifications do not need to be treated similar to Notes within the EOPs (which are expected memory knowledge items) requiring prompt interpretation and action.	Throw out question, inappropriate from memory knowledge item.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 600 psig.	A.1	--------NOTE-------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. -------------------- Declare the associated control rod scram time "slow."	8 hours
	OR		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ATTACHMENT 4

FINAL WRITTEN EXAMINATION AND ANSWER KEY

RIVER BEND EXAM ANSWER KEY
SENIOR REACTOR OPERATOR (01/97)

001	b
002	d
003	b
004	a
005	a
006	b
007	a
008	c
009	b
010	c
011	b
012	c
013	d
014	d
015	~~d~~ c
016	a
017	d
018	c
019	d
020	d

RIVER BEND EXAM ANSWER KEY
SENIOR REACTOR OPERATOR (01/97)

021 b

022 c

023 a

024 a

025 a

026 d

027 a or b

028 c

029 d

030 b

031 b

032 c

033 c

034 ~~b~~ Delete

035 a

036 d

037 b

038 c

039 d

040 a

RIVER BEND EXAM ANSWER KEY
SENIOR REACTOR OPERATOR (01/97)

041 ~~c~~ Delete

042 a

043 c

044 c

045 b

046 b or a

047 c

048 b

049 a

050 b

051 c

052 d

053 d

054 b

055 a

056 c

057 b

058 a

059 ~~d~~ Delete

060 b

RIVER BEND EXAM ANSWER KEY
SENIOR REACTOR OPERATOR (01/97)

061 a

062 d

063 b

064 d

065 b

066 d

067 b

068 b

069 c

070 c

071 c

072 b

073 d

074 b

075 c

076 d

077 a

078 a

079 d

080 b

RIVER BEND EXAM ANSWER KEY
SENIOR REACTOR OPERATOR (01/97)

081 c

082 b

083 b

084 b

085 c

086 c

087 c

088 a

089 a

090 d

091 a

092 a

093 d

094 a

095 c

096 c

097 c

098 b

099 b

100 b

RIVER BEND

SENIOR REACTOR OPERATOR

NRC WRITTEN EXAMINATION

RESTRICTED INFORMATION

OFFICIAL USE ONLY

RIVER BEND SENIOR REACTOR OPERATOR NRC EXAMINATION

FACILITY: River Bend

REACTOR TYPE: BWR-GE6

DATE ADMINISTERED: 01/27/97

CANDIDATE: ______________________________

INSTRUCTIONS TO CANDIDATE:

Use the supplied answer sheet for documentation of your answers. There are 100 multiple choice questions on this examination, each worth 1.00 point. Passing grade for this examination consists of an overall score of 80%. Examination papers will be picked up four (4) hours after the examination begins.

100		
TOTAL POINTS	CANDIDATE'S SCORE	PERCENT

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

____

Candidate's Signature

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination, the following rules apply:

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and on all answer sheet pages.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

9. If the intent of a question is unclear, ask questions of the examiner only.

10. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

1 **QUESTION:**

Following Classification of an Emergency Event, what are the time interval guidelines for making NOTIFICATIONS to State and Local Authorities of the emergency classification?

Notification must be made:

a. immediately.

b. within 15 minutes.

c. within 30 minutes.

d. within 60 minutes.

2 **QUESTION:**

SELECT the conditions under which the Control Room Supervisor (CRS) may concurrently fill the Shift Technical Advisor (STA) position.

The CRS may fill the STA position:

a. provided that he remains in the main Control Room.

b. while in Modes 1, 2 and 3 if the Operations Shift Superintendent remains in the Main Control Room.

c. only when the plant is operating under steady state conditions NOT requiring the Administrative CRS to be stationed.

d. provided that there are a total of five (5) licensed operators currently on shift.

3 **QUESTION:**

Given the following conditions:

- The plant is shutdown for a refueling and maintenance outage
- A Senior Reactor Operator licensed individual has been working more than 12-hours per day during his off-shift days coordinating procedure changes for the outage plant modifications
- He is scheduled to return to shift Saturday at 6:00am as the dayshift Control Room Supervisor

What is the LATEST time he may work on the Friday before he takes the shift as CRS? (Assume that he has not exceeded any overtime restrictions.)

a. 10:00pm

b. 6:00pm

c. 2:00pm

d. 6:00am

4 **QUESTION:**

A currently licensed Senior Reactor Operator (SRO) has started wearing glasses.

Which of the following individuals is responsible for ensuring the proper notifications are made to the NRC regarding modifying the SRO's license?

a. The Senior Reactor Operator

b. The Operations Superintendent

c. The General Manager, Plant Operations

d. The Manager, Training

5 **QUESTION:**

Which of the following permission/notification requirements must be met for an INTENTIONAL entry into Tech Spec 3.0.3?

Permission must be obtained from the:

a. Operations Superintendent and the NRC Resident Inspector notified.

b. General Manager - Plant Operations and the NRC Resident Inspector notified.

c. Manager - Operations and a 1-hour report made to the NRC.

d. General Manager - Plant Operations and a 4-hour report made to the NRC.

6 **QUESTION:**

Given the following conditions:

- Testing is in progress that requires use of a Human Tag
- It is anticipated that the test will require 5 more hours for completion
- Current time is 4:00pm

How much longer may this Human Tag continue?

a. 1 hour

b. 2 hours

c. 4 hours

d. 5 hours

7 **QUESTION:**

Given the following information for a fully qualified, male River Bend radiation worker:

- 1997 Total Effective Dose Equivalent (TEDE) at River Bend as of 05/15/97 is 550 mrem
- This individual spent 5 weeks in March and April of this year (1997) at the Clinton Power Station
- River Bend Radiation Protection has not yet received his radiation exposure information from Clinton

What is the MAXIMUM Total Effective Dose Equivalent (TEDE) that he can receive at River Bend for the remainder of 1997 WITHOUT any additional approvals?

a. 1450 mrem

b. 1950 mrem

c. 2000 mrem

d. 2500 mrem

8 QUESTION:

Given the following conditions:

- The plant is operating at 95% power
- A VALID high reactor pressure condition has just occurred
- The reactor did NOT scram

The At-The-Controls Operator should:

a. inform the Control Room Supervisor of the condition and insert a manual reactor scram when ordered.

b. not insert a manual reactor scram unless the high pressure condition is sustained.

c. insert a manual reactor scram and inform the Control Room Supervisor of the condition and action taken.

d. perform an immediate power reduction to reduce pressure below the high pressure scram setpoint as soon as possible.

9 **QUESTION:**

Given the following conditions:

- The plant is operating at 100% power
- It has just been discovered that a weekly surveillance was not accomplished within its specified frequency.
- It should have been completed three (3) days ago

Which of the following describes the requirements regarding the associated equipment's Operability status?

The equipment:

a. may be considered to be Operable for the surveillance grace period (1.75 days) to allow the surveillance to be performed.

b. may be considered to be Operable for a maximum period of 24 hours to allow the surveillance to be performed.

c. must have a Safety Function Operability Determination performed within 6 hours.

d. must be declared Inoperable immediately but may be returned to service for performance of the surveillance.

10 **QUESTION:**

Given the following conditions:

- The plant is shutdown with all rods fully inserted
- Shutdown cooling is in service with a reactor temperature of 230 °F
- The Operations Shift Superintendent (OSS) has been scheduled to attend a 30 minute outage meeting outside the Main Control Room

Which of the following describes how the control room command function may be maintained during this meeting?

a. The OSS may turnover the command function to the Unit Operator or At-The-Controls Operator.

b. If he is no further than 10 minutes from the Main Control Room the OSS may retain the command function.

c. The OSS must turnover the command function to another individual with a Senior Reactor Operators license.

d. The Administrative Control Room Supervisor (with an inactive license) may assume the command function once the plant is less than 200 °F.

11 **QUESTION:**

EQUIPMENT OUT OF SERVICE signs should be placed next to:

a. automatic initiation pushbutton(s) for an out of service system.

b. controls for equipment required to be maintained operable.

c. equipment that is necessary for operation of out of service system.

d. devices used to isolate the equipment out of service.

12 **QUESTION:**

Given the following conditions:

- Preparations are in progress for a reactor and plant startup
- All control rods are fully inserted
- Reactor coolant temperature is 192 °F
- Both Recirculation Pumps are running in "slow" speed

At what point in the startup process are the control board lineups for safety-related systems (OSP-0017) required to be performed?

The lineups shall be performed:

a. after the Mode Switch is placed in "Startup/Hot Standby" but before control rod withdrawals commence.

b. prior to reaching the point of adding heat with reactor power.

c. prior to reactor coolant temperature exceeding 200 °F

d. at the same time the Administrative Control Room Supervisor is stationed.

13 **QUESTION:**

Following the decision to implement AOP-0031, "Shutdown From Outside The Main Control Room", where must the Operations Shift Superintendent report?

a. Division I Remote Shutdown Panel

b. Division II Remote Shutdown Panel

c. Operations Support Center

d. Technical Support Center

14 QUESTION:

Given the following conditions:

- A loss of coolant accident has occurred
- Fuel damage is suspected but has not yet been confirmed
- The emergency response organization has been activated and the State and local authorities have been notified

Which of the following meets the definition of a "release" for the purposes of offsite notifications?

a. Any monitored plant effluent path is at its high alarm setpoint.

b. Two or more monitored plant effluent paths are at least three (3) times their normal readings.

c. The fuel damage has been confirmed by reactor coolant sampling.

d. A release is detected using radiation monitoring teams with portable monitors.

15 QUESTION:

During performance of flow balance surveillance testing on the Normal Service Water System, several valves that are normally full open need to be throttled for the duration of the test.

Which of the following describes how the change in status of these valve will be tracked?

a. The throttle valve computerized listing will be updated and a Change Notice (CN) for the valve realignment will be completed for SOP-0018, "Normal Service Water".

b. A Procedure Revision for the valve realignment will be completed for SOP-0018, "Normal Service Water" and an entry made in the Control Room log.

c. The repositioning of the valves will be documented per the guidance in ADM-0022, "Conduct of Operations".

d. The repositioning of the valves will be controlled by tracking all manipulations as required in the surveillance test package.

16 QUESTION:

Which of the following describes how an independent verification is performed on a valve that is already "locked open" per a valve lineup?

The independent verifier shall:

a. verify the locking device is in place as required.

b. remove the locking device, verify the valve is fully open and reinstall the locking device.

c. attempt to move the valve in the "closed" direction to verify the locking devic holds the valve open.

d. have another individual verify the position and then verify the locking device installed.

17 QUESTION:

Which of the following controls are required when a freeze seal is being utilized as an isolation point for a scheduled maintenance activity?

a. A Dedicated Operator from Operations shall be stationed at the freeze seal to monitor it and to isolate the system should the seal fail.

b. An Operator Aid Caution Tag shall be posted in the Main Control Room on the system being isolated by the freeze seal.

c. A Danger - Hold tag shall be hung at the freeze seal location prior to the seal being established.

d. The Operations Shift Superintendent shall brief the operating crew on the freeze seal, including contingency actions should the freeze seal fail.

18 **QUESTION:**

The plant is was initially operating at 100% power. A transient occurred resulting in the following conditions:

-RPV level is 36 inches and stable
-Reactor power is 74% and stable
-Total core flow is 50.5 E6 lbm/hr. and stable

The cause of this plant configuration was the receipt of a signal from the:

a. EOC-RPT logic.

b. recirculation pump cavitation interlock circuitry.

c. recirculation flow control valve runback logic.

d. ATWS/ARI logic.

19 **QUESTION:**

Given the following conditions:

- The High Pressure Core Spray (HPCS) system is operating and injecting into the reactor vessel following a valid initiation signal.
- All Narrow Range Reactor Water Level Instrumentation has failed "UPSCALE"

Which of the following describes the status of the HPCS system following the instrument failure?

	HPCS Pump	Injection valve F004	Minimum Flow F012
a.	Running	Open	Closed
b.	Tripped	Open	Closed
c.	Tripped	Closed	Open
d.	Running	Closed	Open

20 **QUESTION:**

Given the following conditions:

- Due to a leak, drywell temperature is increasing
- Conditions have exceeded the Safe Zone of the RPV Saturation Temperature Curve.

Which of the following describes the availability of the Wide Range RPV water level indication?

Wide Range level indication:

a. can be used for determination of actual RPV water level only if the indicator reads above its minimum indicated level.

b. may be used for trending RPV water level changes but not for determination of actual level

c. can be used for trending RPV water levels only if the indicator reads above its specified minimum indicated level.

d. may not be used for either trending or for determination of RPV water level.

21 QUESTION:

Given the following conditions:

- A reactor startup is in progress
- Control rod 16-25 has just been withdrawn ONE notch (Notch 18 to 20)
- The Rod Motion control "Withdraw" pushbutton on the Operator Control Module was used for the withdrawal
- The Rod Control & Information System operated as designed during the withdrawal

SELECT the point in the rod withdrawal sequence when the white "Settle" light would be expected to extinguish.

a. The Rod Motion control "Withdraw" pushbutton has just been released.

b. The hydraulic control unit withdraw exhaust directional control valve has just closed.

c. The white "Out" light on the Rod Motion Matrix has just extinguished.

d. The control rod digital position display on the Core Map Display is indicating "- -".

22 **QUESTION:**

Given the following conditions:

- The plant had been operating at 100% power
- A severe over-pressure transient resulted in ALL Safety Relief Valves (SRV) opening in the "relief" mode and then lifting in their "safety" mode
- RPV pressure peaked at 1200 psig
- All valves have closed (reseated) with the exception of one SRV that remains open in its "safety" mode
- The required actions of AOP-0035, "Safety Relief Valve Stuck Open", have been taken including scramming the reactor but the SRV has not closed.

Which of the following describes the resulting tailpipe temperature trend as the plant cools down and depressurizes through the stuck open SRV? (Assume containment pressure is 0 psig and remains constant.)

SRV tailpipe temperature will:

a. start at 260 °F, rise to approximately 290 °F and then will slowly fall following reactor pressure during the depressurization below 500 psig.

b. start at 525 °F and will slowly fall following reactor pressure during the depressurization.

c. start at 285 °F, rise to approximately 325 °F and then will slowly fall following reactor pressure during the depressurization below 500 psig.

d. start at 305 °F and will slowly fall following reactor pressure during the depressurization.

23 QUESTION

Given the following conditions:

- The plant has experienced a reactor scram and Main Steam Isolation Valve (MSIV) closure from 75% power
- Reactor pressure control is by periodic operation of the Safety Relief Valves (SRV)

Which of the following is an indication that one of the SRV tailpipe vacuum breakers has failed in the "OPEN" position during SRV operation?

a. An increase in indicated suppression pool water level each time the SRV opens.

b. Direct pressurization of the containment each time the SRV opens.

c. Steam bypassing the SRV discharge quenchers with a direct piped path into the suppression pool.

d. Suppression pool water being drawn up into the tailpipe after SRV closure.

24 QUESTION:

Suppression pool temperature is limited to 100°F during normal operations.

Which of the following is an approved exception to this limit?

a. The limit maybe increased to 110°F with the MSIVs closed following a reactor scram (all control rods fully inserted).

b. Suppression pool temperature is allowed to reach 110°F while testing RCIC during normal plant operations.

c. The limit maybe increased above 120°F if reactor pressure is less than 300 psig.

d. Suppression pool temperature is allowed to reach 120°F after a Safety Relief Valve actuation.

25 **QUESTION:**

Given the following conditions:

- The plant is operating at 45% power
- A VALID high steam flow signal is sensed in the "B" Main Steam Line
- Steam flows in the remaining lines are normal

Which of the following describes the expected AUTOMATIC response of the Containment And Reactor Vessel Isolation Control System (CRVICS) to this event?

a. A Group 6 (MSIV & MSL Drains) containment isolation signal will result.

b. Only Channel "B" of the CRVICS MSIV isolation logic will de-energize.

c. One solenoid on each MSIV will de-energize but no valve closures will occur.

d. Only the inboard and outboard MSIVs in the "B" main steam line will close.

26 QUESTION:

Given the following conditions:

- The plant is performing a reactor and plant startup
- Reactor power is 5.5 E4 counts per second (cps) on the Source Range Monitoring (SRM) instrumentation and is increasing

Which of the following describes how a SRM detector that cannot be fully withdrawn affects continued control rod withdrawals? Assume the stuck SRM Channel is NOT bypassed.

Control rod withdrawals will be possible:

a. because the control rod withdrawal block logic is one-out-of-two-taken-twice for SRM detectors.

b. as soon as the three fully withdrawn SRM detector power levels are all less than 100 cps.

c. until power reaches 1.0 E5 cps and then will not be allowed until associated IRM power is at or above Range 3.

d. until power reaches 1.0 E5 cps and then will not be allowed until associated IRM power is at or above Range 8.

27 QUESTION:

Following an actuation of Alternate Rod Insertion (ARI) from 100% power, how can the operator determine if the actuation was "automatic" or "manual"?

The operator should check the:

a. status of the Recirculation Pumps.

b. position of the backup scram valves.

c. final pressure in the scram air header.

d. status of the ARI solenoid valves.

28 **QUESTION:**

The "B" Standby Liquid Control (SLC) Squib Valve (F004B) continuity meter in P642 is reading 4.95 milliamps.

What are the appropriate actions for these conditions?

a. Declare the "B" SLC system Inoperable as this current value indicates the squib ignitor has decomposed.

b. Declare the "B" SLC system Inoperable as this current value indicates the squib valve has fired.

c. Take no action as this is the normal continuity current for the squib valve ignitor circuit.

d. Direct troubleshooting of the "B" SLC system as the current should be no lower than 2 amps.

29 **QUESTION:**

Given the following conditions:

- All four Automatic Depressurization System (ADS) manual initiation pushbuttons have been armed, depressed, and released
- No ADS valves opened

Which of the following describes what must now occur to open the ADS Safety Relief Valves?

a. The 105 second timer must time out.

b. The 105 second timer must time out AND the ADS manual initiation pushbuttons must be depressed again.

c. One low pressure ECCS pump per division must be started.

d. One low pressure ECCS pump must be started AND the ADS manual initiation pushbuttons must be depressed again.

30 QUESTION:

Given the following conditions:

- LPCS System Inoperative alarm has been received
- The Low Pressure Core Spray (LPCS) Line Break status light (postage stamp) is illuminated

Which of the following describes the location of the break?

The break is in the:

a. Drywell between the LPCS Testable Check Valve (F006) and the LPCS Injection Valve (F005).

b. reactor pressure vessel downcomer area.

c. Auxiliary Building between the LPCS Testable Check Valve (F006) and the LPCS Injection Valve (F005).

d. area inside the core shroud bypassing the Low Pressure Core Spray sparger.

31 **QUESTION:**

Given the following conditions:

- The Low Pressure Core Spray (LPCS) system is running in the test return to the suppression pool mode
- A leak has caused drywell pressure to increase to 1.95 psig
- Reactor water level is -62 inches
- Reactor pressure is 750 psig

Select the expected AUTOMATIC response of the LPCS system.

a. The LPCS Pump trips, the Test Return Valve To Suppression Pool (F012) closes, the pump restarts and the Injection Isolation Valve (F005) opens.

b. The Test Return Valve To Suppression Pool (F012) closes and the LPCS Pump continues to run on minimum flow.

c. The LPCS Pump trips, the Test Return Valve To Suppression Pool (F012) closes, the pump restarts and runs on minimum flow.

d. The LPCS Pump continues to run, the Test Return Valve To Suppression Pool (F012) closes and the Injection Isolation Valve (F005) opens.

32 **QUESTION:**

Which of the following describes how the Residual Heat Removal (RHR) Heat Exchanger Bypass Valves (F048A & B) should be positioned after an ECCS initiation signal?

F048A & B should be:

a. overridden and closed if suppression pool temperature reaches 100 °F and EOP-2, "Primary Containment Control" is entered.

b. overridden and closed once full injection flow has been established.

c. closed as directed by the Emergency Operating Procedures 10 minutes after the initiation signal.

d. left open to provide maximum Low Pressure Coolant Injection (LPCI) flow.

33 QUESTION:

Which of the items below describes the response of the Rod Control and Information System when the DRIFT TEST pushbutton is depressed at the Operator Control Module (OCM)?

When the DRIFT TEST pushbutton is depressed:

a. red LED drift lights will lite for all rods not at a latched position.

b. a drift signal will be generated for the selected control rod.

c. an artificial drift condition will be induced when the selected rod is moved.

d. the drift circuitry will be verified operable by initiating a drift alarm.

34 QUESTION:

Given the following conditions:

- The HPCS Diesel Generator had been in a normal standby lineup
- An automatic start signal has just been received
- The diesel generator has responded as designed
- It has been determined that NO VALID start signal condition exists

What are the required IMMEDIATE actions for these conditions?

a. Depress the local panel Emergency Stop pushbutton and place the Engine Mode Control Switch in "TEST".

b. Depress the Main Control Room Emergency Stop pushbutton and direct the local operator to place the Engine Mode Control switch in "MAINT".

c. Place the Diesel Generator Mode Switch to "LOCAL MANUAL" and direct the local operator to place the Diesel Generator Control switch to "STOP".

d. Verify the Diesel Generator Mode Switch in "REMOTE MANUAL" and place the Main Control Room Diesel Generator Control switch to "STOP".

35 **QUESTION:**

Given the following conditions:

- A 4160 V ITE type breaker was racked in following maintenance on its associated pump
- The breaker is currently closed
- A loss of DC control power to that breaker occurs
- No operator actions are taken

Which of the following describes the operational capabilities of this breaker?

a. The breaker cannot be remotely operated but can be locally tripped, then closed and tripped open one more time.

b. The breaker will trip open on loss of control power and all additional breaker operations must be performed locally.

c. The breaker will trip open on loss of control power and no further breaker operations are possible.

d. The breaker cannot be remotely operated but can be locally tripped one time with no further operation possible.

36 **QUESTION:**

Given the following conditions:

- The plant is operating at 95% power
- The "B" EHC Pressure Regulator is in "TEST"
- The "A" EHC Pressure Regulator output has failed "AS-IS"
- The At-The-Controls Operator begins a power reduction using recirculation flow

Which of the following would be the expected response of EHC for this failure?

a. The Maximum Combined Flow Limiter will limit the magnitude of the pressure error signal generated as power decreases.

b. The Turbine Control Valves will open attempting to maintain the selected pressure setpoint as reactor power and pressure decrease.

c. The Maximum Combined Flow Limiter will close the Main Turbine Bypass Valves to control pressure at the setpoint.

d. The Turbine Control Valves will not move, however, turbine load will decrease as reactor pressure follows the power reduction

37 **QUESTION:**

Which of the following plant systems is required to be Operable to support the Operability requirements of Secondary Containment?

a. Combustible Gas Control

b. Standby Gas Treatment System

c. Penetration Valve And Main Steam Isolation Valve-Leakage Control Systems

d. Leak Detection System

38 **QUESTION:**

The Average Power Range Monitoring (APRM) Gain Adjustment Factor (GAF) value for APRM Channel "B" is 0.989, with Core Thermal Power at 100%.

Select the current condition of this APRM channel and the appropriate actions for that condition.

a. Actual power is greater than indicated power and a GAF adjustment is required.

b. A conservative condition exists but a GAF adjustment is required.

c. Indicated reactor power is greater than actual power and a GAF adjustment is NOT required.

d. A non-conservative condition exists but GAF adjustment is NOT required.

39 **QUESTION:**

Given the following conditions:

- The High Pressure Core Spray (HPCS) system is running in the CST to CST mode
- While the pump is running an I&C Technician's error results in a low CST water level signal

Which of the following PREVENTS pumping suppression pool water into the CST for these conditions?

a. The HPCS Pump will trip as soon as the Suppression Pool Suction Valve (F015) begins to open with HPCS in the CST to CST mode.

b. The HPCS low CST water level automatic swap to the suppression pool is overridden with HPCS in the CST to CST mode.

c. The operator must take immediate action to close the Test Bypass Valve To CST (F010) and Test Return Valve To CST (F011).

d. The Test Bypass Valve To CST (F010) and Test Return Valve To CST (F011) both close as the Suppression Pool Suction Valve (F015) begins to open.

40 **QUESTION:**

A turbine load set runback will NOT occur on a load reject if the plant is operating at or below 40%.

SELECT the parameter used as input to the load reject circuitry for the determination of plant power.

a. High pressure turbine exhaust pressure.

b. High pressure turbine first stage pressure.

c. Main generator power output (MWe).

d. Main generator current output (amps).

41 **QUESTION:**

Prior to a startup with the plant in a cold, depressurized condition, the Unit Operator adjusts the Control Rod Drive Hydraulic (CRD) system Pressure Control Valve (PCV) to maintain 250 psid between drive water header pressure and reactor pressure.

How is this pressure differential maintained as reactor pressure increases during the ensuing startup to 100% power? (Assume the CRD system is operating as designed.)

As reactor pressure increases during the startup:

a. the Pressure Control Valve automatically operates to maintain CRD system pressure above reactor pressure.

b. the Unit Operator must periodically open the Pressure Control Valve to maintain the required differential pressure.

c. the Flow Control Valve automatically opens to maintain a constant flow, therefore a constant d/p across the PCV.

d. the Unit Operator must periodically adjust the Flow Control Valve to maintain CRD system pressure above reactor pressure.

42 **QUESTION:**

Given the following conditions:

- The plant is performing a scheduled shutdown
- Intermediate Range Monitoring (IRM) channel "H" has failed "UPSCALE" and has NOT been bypassed

At what point would an automatic half scram be expected for these conditions?

a. The plant enters Mode 2.

b. APRM "H" reaches 5% power.

c. The IRM detectors are fully inserted.

d. Power has decreased to the Low Power Setpoint.

43 **QUESTION:**

Which of the following would be an indication of a DECREASE in the efficiency of the Offgas system catalytic recombiners?

a. Main Condenser vacuum will decrease.

b. Offgas recombiner temperatures will increase.

c. Offgas system total flow volume will increase.

d. Offgas system discharge radiation levels due to Nitrogen-16 will decrease.

44 **QUESTION:**

Given the following conditions:

- The plant is operating at 50% power
- The Main Seal Oil Pump has just tripped
- The Emergency Seal Oil Pump did not start

Which of the following describes the required actions for the above conditions?

a. An immediate generator and turbine trip is required.

b. A rapid power reduction to less than 25% is required.

c. Begin load reduction for main generator purge.

d. Hydrogen pressure should be reduced to approximately 40 psig.

45 **QUESTION:**

The following is the current Rod Control & Information System display data for control rod 28-25 that had been at Notch "48". (Note that the LEDs will illuminate/not illuminate when the appropriate switches are depressed, as required.)

- Full In: Illuminated
- Full Out: NOT Illuminated
- Drift: Illuminated
- Selected: NOT Illuminated
- Accum Fault: NOT Illuminated
- Scram: NOT Illuminated

Which of the following has occurred?

a. The control rod was driven to Notch "00" using the "Insert" pushbutton.

b. The Scram Outlet Valve (127) ONLY opened.

c. The Scram Inlet Valve (126) ONLY opened.

d. BOTH the Scram Inlet (126) and Outlet Valves (127) have opened.

46 QUESTION:

Which of the following is indication that a failure of both seals on one recirculation pump have occurred?

a. Drywell identified leakage will decrease and unidentified leakage will increase.

b. Drywell identified leakage will increase and unidentified leakage will increase.

c. Drywell identified leakage will increase and unidentified leakage will decrease.

d. Drywell identified leakage will decrease and unidentified leakage will decrease.

47 QUESTION:

Given the following conditions:

- The plant is performing a reactor startup and heatup
- Reactor water level control is via the Reactor Water Cleanup (RWCU) system rejecting to the main condenser
- Main condenser vacuum has been established with the Mechanical Vacuum Pumps

Which of the following is the reason for limiting reject flowrate for these conditions?

Discharging water to the main condenser at a vacuum:

a. with excessive flowrates can exceed the design capacity of the mechanical vacuum pumps.

b. may cause a RWCU pump trip due to high flowrates through the system.

c. can exceed the Reactor Plant Component Cooling Water system heat removal capabilities for the non-regenerative heat exchanger.

d. will cause damage to the filter-demin resin strainers and resin "channeling" from high flowrates.

48 QUESTION:

Given the following conditions:

- A plant startup is in progress
- The main turbine is rolling at 1050 rpm with "1500" rpm selected at the "medium" starting rate
- The At-The-Controls Operator reports the acceleration rate of the turbine is approximately 180 rpm

Which of the following describes the current main turbine/EHC operation?

a. A loss of both EHC speed signals has occurred.

b. A loss of one of the EHC speed signals has occurred.

c. The turbine is accelerating at the appropriate rate as selected by the speed control unit.

d. The wobulator circuit is starting to vary acceleration rate as turbine speed approaches 1500 rpm.

49 **QUESTION:**

Given the following conditions:

- The plant is operating at 50% power
- Main Steam Isolation Valve (MSIV) partial stroke testing is in progress
- The Inboard MSIV on the "D" Main Steam Line (MSL) is being tested with the "TEST" pushbutton
- While the MSIV is stroking in the closed direction, power is lost to the test solenoid

Which of the following describes how this MSIV will be affected?

The Inboard MSIV on the "D" MSL:

a. will immediately reopen with no operator action.

b. can be reopened with the normal control switch.

c. will reopen over 4 minutes as the air bleeds off the top of the operating piston.

d. will fully close and cannot be reopened until power is restored.

50 **QUESTION:**

The RHR S/D Cooling Isolation Valve Enable/Disable switch on the local panel (P001) has two positions, "Enable/Disable".

Which of the following describes when the switch is REQUIRED to be in "Disable" and the effect on the operation of Shutdown Cooling when it is in this position?

The RHR S/D Cooling Isolation Valve Enable/Disable switch is placed in "Disable" when:

a. reactor pressure is greater than 135 psig and prevents operation of the RHR Shutdown Cooling Inboard Isolation Valve (F009) from the Main Control Room.

b. reactor pressure is greater than 135 psig and prevents operation of the RHR Shutdown Cooling Outboard Isolation Valve (F008) from the Main Control Room.

c. evacuating the Main Control Room to allow local operation of the RHR Shutdown Cooling Inboard Isolation Valve (F009).

d. evacuating the Main Control Room to allow local operation of the RHR Shutdown Cooling Outboard Isolation Valve (F008).

51 **QUESTION:**

Which of the following is normally cooled by the Reactor Plant Component Cooling Water system on the Recirculation Pump shaft seal packages?

a. The reactor coolant leaking from the Recirculation System into the lower seal cavity via the breakdown bushing.

b. The reactor coolant flow directed to the Drywell Equipment Drain Sump from the seal package.

c. The Control Rod Drive Hydraulic System seal purge flow being injected into the Recirculation Pump impeller cavity.

d. The Control Rod Drive Hydraulic System seal purge flow being directed to the Drywell Equipment Drain sump.

52 **QUESTION:**

Given the following conditions:

- The plant is operating at 85% power
- The At-The-Controls Operator has just depressed the "Transfer To LFMG" pushbuttons for transferring the Recirculation Pumps to "slow" speed
- The Operator reports that the CB5A breaker opened but the CB5B breaker did NOT open

Which of the following describes the expected status of the Recirculation Pumps 20 seconds after this failure?

a. Both Recirc Pumps will be coasting to a stop

b. The "A" Recirc Pump will be running in "slow" speed, the "B" Recirc Pump will be running in "fast" speed.

c. The "A" Recirc Pump will be running in "slow" speed, the "B" Recirc Pump will be coasting to a stop.

d. The "A" Recirc Pump will be coasting to a stop, the "B" Recirc Pump will be running in "fast" speed.

53 **QUESTION:**

Which of the following describes how the design and operation of the two backup scram valves results in venting the scram air header?

a. Normally energized, one valve will deenergize with each RPS channel to vent the scram air header.

b. One valve is powered from each RPS trip channel with either valve energizing to vent the scram air header.

c. Both RPS channels must trip to deenergize either valve with both valves required to vent the scram air header.

d. Both RPS channels must trip to energize either valve with only one valve required to vent the scram air header.

54 **QUESTION:**

SELECT the method by which reactivity insertion rate is regulated for control rod withdrawals and insertions.

Reactivity insertion rate is controlled by:

a. automatically varying the position of the Control Rod Drive Hydraulic Pressure Control Valve.

b. throttling the water flow entering and leaving the under-piston area of the drive mechanism.

c. automatically varying the operating position of the Control Rod Drive Hydraulic Flow Control Valve.

d. throttling the water flow entering and leaving the over-piston area of the drive mechanism.

55 **QUESTION**

One of the concerns with maintaining proper reactor water level in the steam separators during plant operation is to minimize "carryunder".

Which of the following would result if excessive "carryunder" were occurring?

a. Core thermal power would decrease.

b. Core differential pressure would decrease.

c. Overall plant efficiency would increase.

d. Jet pump net positive suction head would increase.

56 QUESTION

Given the following conditions:

- The plant is shutdown for refueling
- The "A" Residual Heat Removal (RHR) loop is being placed in the Fuel Pool Cooling Assist mode on the containment (upper) pools

Which of the following prevents draining the upper containment pools to the suppression pool via the RHR Minimum Flow To Suppression Pool Valve (F064A) when starting the "A" RHR Pump?

a. The RHR pump will trip if a flowpath to the upper pools is not established within 8 seconds.

b. F064A is overridden closed by the operator during the start of the RHR pump.

c. The operator must establish flow to the upper pools before the F064A valve automatically opens.

d. The F064A valve is isolated and tagged closed during the Fuel Pool Cooling Assist lineup.

57 **QUESTION:**

A loss of power has just occurred to one of the solenoids for an Main Steam Isolation Valve (MSIV).

Which of the following describes the response of the MSIV and the reason for that response?

The MSIV will:

a. close because the solenoids energize to align the air supply to open the MSIV.

b. remain open because the other solenoid continues to supply air to the MSIV.

c. close because the solenoids are in series and either one deenergizing will vent the air supply to the MSIV.

d. remain open because the instrument air accumulator for that MSIV continues to supply air to the actuator.

58 **QUESTION:**

Given the following conditions:

- The plant is operating at 100% power
- Feedwater level control is in "automatic" with Narrow Range Level "A" selected
- The "A" Narrow Range channel has just failed "DOWNSCALE"
- No operator actions are taken

SELECT the cause of the subsequent reactor scram.

a. Reactor vessel high water level

b. Main Steam Isolation Valve closure

c. Reactor vessel low water level

d. APRM high thermal power

59 **QUESTION:**

Following a normal reactor scram, at what point must the At-The-Controls Operator be alert for narrow range reactor water level indication "outgassing" or "notching"? (Assume the Reference Leg Backfill system is NOT in service.)

a. If the cooldown rates exceed the Technical Specification limits.

b. If the MSIVs close and pressure control is on the Safety Relief Valves.

c. When the last Feed Pump is removed from service.

d. As reactor pressure decreases below 450 psig.

60 **QUESTION:**

Given the following conditions:

- The plant is shutdown for refueling
- The Inclined Fuel Transfer System (IFTS) cart has just been loaded with two fuel bundles and is still in the upper fuel transfer pool
- A steadily decreasing water level is occurring in the refueling cavity

Which of the following describes what should be done with the fuel on the IFTS cart in order to place it in a safe position?

a. The cart should be left where it is with a source of makeup water supplied to the transfer tube within 30 minutes.

b. The cart should be sent to the lower fuel transfer pool.

c. Open the IFTS Flapper Valve such that makeup water to the upper fuel transfer pool will be supplied to the cart.

d. The fuel should be removed and placed in the upper fuel pool storage racks.

61 QUESTION:

Given the following conditions:

- The plant had been operating at 100% power for several days
- A combination of problems has resulted in the loss of ALL high pressure feed capabilities with the exception of the Reactor Core Isolation Cooling (RCIC) system which automatically initiated
- It has been 2 minutes since the reactor scrammed
- Pressure control is on the main turbine bypass valves

Which of the following describes the proper response of RCIC for these conditions?

RCIC is running with a discharge pressure of:

a. 1010 psig, pump flow of 600 gpm and reactor water level is decreasing.

b. 1030 psig, pump flow of 400 gpm and reactor water level is rising.

c. 940 psig, pump flow of 120 gpm and the Minimum Flow To Suppression Pool valve (F019) is open.

d. 1200 psig, turbine speed of 4250 rpm and the Minimum Flow To Suppression Pool valve (F019) is open.

62 **QUESTION:**

Given the following conditions:

- The Main Control Room has been evacuated due to a fire
- All immediate actions were attempted prior to leaving the Main Control Room however the shutdown status of the reactor could not be determined prior to evacuation
- Control of the plant has been established at the Remote Shutdown Panel

Which of the following must the Control Room Supervisor (CRS) use to make the decision to enter EOP-1A, "RPV Control - ATWS" for these conditions?

The decision will be based upon:

a. the Reactor Building Operator's report on the status of the Scram Inlet and Outlet Valves on the HCUs.

b. monitoring reactor water level trend versus the number and type of injection systems operating.

c. the rate at which Suppression Pool temperature rises due to Safety Relief Valve cycling.

d. an estimate of reactor power from the number of Safety Relief Valves cycling versus time after the scram.

63 **QUESTION:**

Given the following conditions:

- The plant had been operating at 75% power
- A loss of feedwater occurred
- The Main Steam Isolation Valves closed on a level spike below -143 inches
- Reactor water level decreased to -50 inches and is now +30 inches
- All expected automatic actions occurred

Upon resetting the scram, the At-The-Controls Operator received all eight (8) RPS Scram SOV lights, however, the Scram Discharge Volume (SDV) Vent and Drain Valves do NOT indicate "open".

SELECT the reason for the SDV Vent and Drain valves remaining closed.

a. The SDV high water level scram signal is still present.

b. The Alternate Rod Insertion Valves are still open.

c. The Reactor Mode Switch is still in "Shutdown".

d. The EOP-0005, Enclosure 12 (Defeating RPS & ARI Logic Trips) switches are still in "Emergency".

64 QUESTION:

Given the following conditions:

- The plant is operating at 30% power
- Suppression pool cooling is in service
- The surveillance test for manual operation of the Safety Relief Valves (SRV) is in progress
- During the surveillance, suppression pool temperature reached 103 °F

Which of the following are the requirements concerning entry into/implementation of EOP-2, "Primary Containment Control"?

a. The SRV surveillance procedures allow 4 hours to reduce suppression pool temperature below 100 °F before EOP-2 entry is required.

b. EOP-2 may be deferred for 24 hours while suppression pool temperature is reduced to less than 100 °F.

c. Technical Specifications modify the Emergency Operating Procedure limit to 105 °F while surveillance testing to the suppression pool is occurring.

d. The actions of EOP-2 are required to be performed as soon as suppression pool temperature is above 100 °F.

65 QUESTION:

Injecting boron before exceeding the Boron Injection Initiation Temperature (BIIT) will ensure that the suppression pool temperature will not:

a. exceed 160°F.

b. exceed the Heat Capacity Temperature Limit (HCTL).

c. reach the point of adversely affecting low pressure ECCS pump net positive suction head.

d. be heated to the point that drywell pressure increases solely due to high suppression pool temperatures.

66 **QUESTION:**

Given the following conditions:

- The plant is operating at 100% power
- Offgas Building area radiation levels are rising
- Offgas Building and Main Plant Vent Stack effluent radiation levels are rising
- Offgas hydrogen concentration is 5%
- Actions are being directed to reduce hydrogen concentration in Offgas to less than 4%

Which of the following is the reason why the operator is NOT allowed to place the standby recombiner in service to reduce hydrogen concentrations to less than 4%?

Placing the standby Offgas recombiner in service:

a. may exceed the capacity of the preheater reducing temperatures to the level where recombination will stop.

b. could reduce the flow rate when the service air purge is initiated.

c. may cause a loss of main condenser vacuum resulting in an MSIV closure if the plant is still at power.

d. could provide the ignition source for the hydrogen already present.

67 **QUESTION:**

Given the following conditions:

- The plant had been operating at 100% power
- A loss of off-site power with diesels failing to start (Station Blackout) has occurred
- All automatic actions have occurred and all immediate operator actions have been taken

Which of the following is the concern with long term operation of the Reactor Core Isolation Cooling (RCIC) system under these conditions?

RCIC operation during a station blackout will be limited:

a. by the system isolation on equipment space high temperatures.

b. by the cooling water being supplied to the RCIC lube oil cooler.

c. by turbine exhaust backpressure as the suppression pool reaches saturation conditions.

d. by the automatic isolation on low steam supply pressure as the reactor depressurizes.

68 QUESTION:

Given the following conditions:

- A reactor scram with Main Steam Isolation Valve (MSIV) closure from 90% power has occurred
- ADS air is not available to the Safety Relief Valves (SRV)
- EOP-1, "RPV Control" requires maintaining the cooldown rate less than 100 °F/hr
- The EOP directs depressurization by "sustained" SRV opening

Which of the following is the direction the Control Room Supervisor (CRS) should provide to the Unit Operator regarding "sustained" SRV opening?

The SRVs should be:

a. opened and maintained open until the reactor is depressurized or until air pressure is no longer available.

b. opened and maintained open to obtain pressure reductions equivalent to 100 °F before shutting.

c. opened to obtain a rapid pressure reduction to 450 psig and then placed in "Automatic".

d. opened to obtain pressure reductions in increments of 100 psig until the reactor is depressurized.

69 **QUESTION:**

The bases for the Maximum Core Uncovery Time Limit curve:

a. is the time limitation for a reactor water level indication response once all injection is stopped and adequate core cooling is assured.

b. is the time the core may be partially covered and steam cooling ensures the hottest fuel rod temperature will not exceed 1800°F.

c. is the time reactor level may be below the bottom of active fuel with no cooling and the hottest fuel rod clad temperature not exceeding 1500°F

d. provides time limitations on the operator to complete RPV flooding prior to exiting this procedure and entering EOP-4, "Primary Containment Flooding".

70 **QUESTION:**

Assuming the affected control rods are not physically stuck, which of the following Alternate Control Rod Insertion methods will be most effective for rod insertion without regard to the cause of a failure to scram (ATWS)?

a. Deenergize the scram solenoids

b. Vent the scram air header

c. Vent the CRD overpiston volume

d. Reset the scram and initiate a manual scram

71 **QUESTION:**

The plant is performing an Emergency Depressurization from normal operating pressure. Only three (3) Safety Relief Valves (SRV) can be opened.

Which of the following restrictions/limitations control the opening of the Main Steam Isolation Valves (MSIV) to assist in the depressurization?

The MSIVs:

a. should not be opened if the Tech Spec cooldown rate limits will be exceeded.

b. should not be opened if the cause of the isolation was high steam line radiation levels.

c. may be opened irrespective of any suspected fuel failure or steam line break.

d. may be opened only if the differential pressure across the valves can be reduced to less than 50 psid.

72 **QUESTION:**

Given the following conditions:

- The plant had been operating at 100% power
- The "A" Heater Drain Pump has tripped and the "B" Heater Drain Pump is NOT available
- Reactor power is slowly increasing

SELECT the power reduction REQUIRED for these conditions.

Power should be reduced to:

a. 70%.

b. 75%.

c. 80%.

d. 90%.

73 **QUESTION:**

Given the following conditions:

- A loss of coolant accident has occurred
- All plant systems have responded as designed
- Conditions are such that Emergency Containment Venting is required by EOP-2, "Primary Containment Control"

Which of the following systems is REQUIRED to be available to perform this venting?

a. Annulus Pressure Control

b. Drywell/Containment Purge Supply

c. Hydrogen Mixing

d. Hydrogen Purge

74 **QUESTION:**

SELECT the reason why terminating and preventing injection during a failure-to-scram (ATWS) transient results in a power reduction.

Terminating and preventing injection:

a. results in increased core inlet subcooling as feed preheating is decreased.

b. increases the void fraction by a reduction in core natural circulation flow.

c. increases water temperature as level is decreased.

d. results in an increase in fuel temperatures as core steaming rate increases.

75 **QUESTION:**

During a loss of coolant accident, containment pressure reached 35 psig.

Which of the following describes the concern with a containment pressure of this magnitude?

a. The containment "burst" pressure was exceeded.

b. The Safety Relief Valves cannot be opened at this pressure.

c. The containment venting capability is not available.

d. Suppression pool/containment water level indication is no longer reliable.

76 **QUESTION:**

Which of the following is a consequence of allowing suppression pool water level to decrease below 13 feet?

Suppression pool water level less than 13 feet:

a. uncovers the Reactor Core Isolation Cooling turbine exhaust line.

b. reduces the available net positive suction head for the low pressure ECCS pumps below minimum required.

c. uncovers the top two horizontal vents.

d. could result in overpressurization of the Containment.

77 **QUESTION:**

Given the following conditions:

- A reactor startup is in progress
- The At-The-Controls Operator is withdrawing control rods approaching criticality
- A "SRM Short Period" alarm is received during control rod withdrawal.

What are the REQUIRED actions for these conditions?

The At-The-Controls Operator should immediately:

a. stop control rod withdrawal and allow the period to stabilize.

b. insert a reactor scram if reactor period is less than 30 seconds.

c. request direction from the Reactor Engineer/Control Room Supervisor.

d. insert the control rod fully and notify the Control Room Supervisor.

78 **QUESTION:**

Given the following conditions:

- The plant has had a failure-to-scram (ATWS) and has deliberately lowered water level
- Reactor water level is -100 inches
- Reactor pressure is 385 psig
- Suppression pool water level is 19 feet 9 inches
- The Control Room Supervisor has just directed the operators to commence and slowly inject with the systems listed in EOP-4A, "ATWS - Level/Power Control"

Which one of the EOP-4A listed systems must have interlocks defeated prior to use as an injection system?

a. Low Pressure Coolant Injection

b. Reactor Core Isolation Cooling

c. High Pressure Core Spray

d. Condensate/Feedwater

79 **QUESTION:**

Given the following conditions:

- The plant is shutdown with refueling operations in progress
- The water level in the refueling cavity is decreasing due to a leak

Which of the following describes the level indicator used and the conditions under which the Main Control Room will see this level decrease?

The refueling cavity level decrease will:

a. be seen immediately on the Wide Range level indication (recorder and meter).

b. indicate on the Narrow Range (recorder and meter) as soon as level reaches the top of the main steam lines.

c. be seen immediately on the Upset Range level indication (recorder).

d. indicate on the Shutdown Range (meter) before level reaches the reactor vessel flange.

80 **QUESTION:**

Given the following conditions:

- A plant startup is in progress with the Reactor Mode Switch in "Startup/Hot Standby"
- The Outboard Main Steam Isolation Valves have just been opened and steam line warming is in progress
- Main condenser vacuum has been established
- A loss of the "A" RPS Bus has just occurred

How will this bus loss affect the plant assuming it has NOT been restored as directed by AOP-0010, "Loss Of One RPS Bus"?

a. The Outboard Main Steam Isolation Valves will begin to drift close.

b. Main condenser vacuum will begin to decrease.

c. The Recirculation Pumps will automatically trip.

d. The Scram Discharge Volume will begin filling.

81 QUESTION:

Following a transient, the Containment Ventilation system is aligned and operating as follows:

- The "A" and "B" Unit Coolers are running, the "C" Unit Cooler is off
- The Containment Chilled Water Supply and Return Containment Isolation Valves are closed
- The Standby Service Water supply valves to the "A" and "B" unit coolers are closed

Which of the following describes what has occurred?

a. Drywell pressure has been 3.0 psig for 6 minutes.

b. A loss of offsite power has occurred and all three diesel engines started 3 minutes ago.

c. Reactor water level has decreased to -60 inches.

d. Containment to annulus D/P decreased to -14" WG (vacuum in the containment)

82 QUESTION:

Given the following conditions:

- The Main Control Room has been evacuated due to toxic gas
- ALL immediate actions were accomplished prior to leaving
- The Reactor Core Isolation Cooling (RCIC) was operating for reactor water level control
- The RCIC turbine coasted to a stop

Which of the following describes what occurred to the RCIC system and the system's current status?

The RCIC:

a. system has isolated and is no longer available for reactor water level control from the RSP.

b. turbine tripped, can be locally reset and then restarted from the RSP.

c. system has isolated and may be restored once the isolation signal is reset from the RSP.

d. turbine tripped, can be both reset and restarted from the RSP.

83 QUESTION:

While operating in EOP-3, "Radioactive Release Control", the operator is directed to restart Turbine Building Ventilation if it is shutdown.

Which of the following describes how this will affect the Turbine Building and the release that is occurring?

Restarting Turbine Building Ventilation will:

a. ensure all building air will be filtered prior to release to the environment.

b. assure overall radioactive releases will be monitored.

c. maintain a positive pressure inside the building.

d. improve building equipment continued operability.

84 QUESTION:

Given the following conditions:

- Prior to the transient the plant had been at 100% power
- A reactor scram and Recirculation Pump automatic transfer to "slow" speed occurred

Which of the following conditions DIRECTLY caused the scram AND Recirculation Pump transfer?

a. Reactor vessel pressure 1127 psig

b. Main Turbine Stop Valve closure

c. High drywell pressure

d. Reactor vessel water level (Level 4)

85 **QUESTION:**

SELECT the reactor vessel water level indicator that provides the most accurate level indication to determine if main steam line flooding is occurring during a severe overfeeding transient?

a. Narrow range

b. Wide range

c. Upset range

d. Shutdown range

86 **QUESTION:**

Which of the following conditions will result in a Control Rod Drive (CRD) mechanism high temperature?

a. Two CRD Stabilizing Valves fail closed.

b. Loss of power to the CRD Pressure Control Valve.

c. Loss of air to the inservice CRD Flow Control Valve.

d. The cooling/exhaust header pressure equalizing valves fail closed.

87 **QUESTION:**

Given the following conditions:

- The plant is operating at 75% power
- The Steam Sea! Evaporator has just been lost
- There is NC . ie estimate for return of the evaporator

SELECT the appropriate operator actions for the above conditions.

a. Reduce turbine load as necessary to maintain the self-sealing steam supply to the turbine glands.

b. Transfer the Recirculation Pumps to "slow" speed and maintain power within bypass valve capacity.

c. Reduce power as required to prevent condenser vacuum from decreasing to less than 25 in Hg.

d. Transfer the Recirculation Pumps to "slow" speed and then trip the main turbine.

88 **QUESTION:**

Given the following conditions:

- The plant has just had a loss of shutdown cooling
- Actions were taken to raise water level to +75 inches as indicated on the SHUTDOWN Range Level Instrument
- No other means of forced circulation is immediately available

What will be the concern if water level subsequently decreases BELOW +75 inches?

Lowering water level:

a. may adversely affect the validity of the reactor coolant temperature indications.

b. will reduce the required net positive suction head for restarting shutdown cooling.

c. reduces the margin to the low water level shutdown cooling isolation setpoint.

d. reduces the amount of relatively cool water available to control the reactor temperature increase.

89 **QUESTION:**

Given the following conditions:

- A reactor and plant startup is in progress with power at 2%
- RPV pressure is 800 psig
- The Reactor Mode Switch is in "Startup/Hot Standby"
- The "A" CRD Pump has just tripped
- The "B" CRD Pump is being lined up for a start
- Scram accumulator 32-41 trouble alarm is in and has been verified to be from low pressure (indicating 1500 psig)

Which of the following will the Control Room Supervisor (CRS) use to make the determination on declaring control rod 32-41 "Slow" or "Inoperable"? (Note: No other accumulator trouble alarms are in.)

The determination is based upon:

a. the control rods' last scram time surveillance results.

b. the time charging water header pressure will be less than 1520 psig.

c. the number of control rods fully withdrawn.

d. drive water pressure availability to fully insert the control rod.

90 QUESTION:

On a total loss of Normal Service Water (NSW) which of the following conditions REQUIRE a manual reactor scram?

a. Service water temperatures are greater than 90 °F and only one NSW Pump can be started.

b. The Standby Service Water system did not start and align itself to the NSW system correctly.

c. The reactor is critical at 175 °F, just below the point of adding heat, with power ascension in progress.

d. The plant is performing a startup and is just making preparations to roll the main turbine.

91 QUESTION:

Given the following conditions:

- The plant is operating at 75% power
- A loss of instrument air has occurred
- Air pressure has reached 67 psig and is steady
- No control rods have drifted
- All immediate actions have been taken
- The At-The-Controls Operator depresses the Feedwater Regulating Valve reset pushbuttons

What will be the expected result for this action?

Reactor water level will:

a. rapidly increase.

b. rapidly decrease.

c. remain at the current level.

d. return to the Master Level Controller setpoint.

92 QUESTION:

Given the following conditions:

- The plant was operating with both Recirculation Pumps in "Slow" speed
- The "A" Recirculation Pump has tripped
- Nuclear instrumentation indicates NO power oscillations are occurring
- The plant is operating in Region "B" of the Power/Flow Map

SELECT the action REQUIRED to exit Region "B".

a. Insert control rods using the Shutdown Control Rod Sequence Package.

b. Restart the "A" Recirculation Pump in "Slow" speed.

c. Transfer the "B" Recirculation Pump to "Fast" speed.

d. Reduce flow in the "B" Recirculation loop.

93 QUESTION:

Given the following conditions:

- The plant was operating at 75% power
- A Group VI (MSIV) and Group VII (RWCU) isolation occurred

Which of the following signals could have caused these isolation?

a. Low reactor vessel water level (Level 2)

b. High drywell pressure

c. RWCU equipment area high temperature

d. Main steam line tunnel high temperature

94 **QUESTION:**

Given the following conditions:

- Drywell temperature and pressure are increasing due to a leak
- Drywell pressure is 1.82 psig
- All expected automatic actions have occurred
- EOP-2, "Primary Containment Control", was entered for high drywell temperature and Enclosure 20 for restoration of drywell cooling is in progress

Once the interlocks have been defeated, which of the following will be providing cooling to the drywell?

a. Drywell cooler fans and normal service water.

b. Drywell cooler fans and chilled water.

c. Drywell cooler fans and standby service water.

d. Drywell cooler fans ONLY.

95 **QUESTION:**

SELECT the MINIMUM conditions assuring adequate core cooling exists while performing Primary Containment Flooding per EOP-4.

Containment water level is:

a. 65 feet with the RPV at 51 psig.

b. above the Maximum Containment Water Level Limit with the containment vented.

c. 65 feet with the RPV vented.

d. above the Maximum Containment Water Level Limit with the containment at 25 psig.

96 **QUESTION:**

Given the following conditions:

- The plant is operating in EOP-2, "Primary Containment Control" for high suppression pool level.

Which of the following is the concern with a continued increase in suppression pool level greater than 21 feet?

a. Suppression pool temperature indications will not be reliable.

b. ECCS pump suction line pressure and loading limitations may be exceeded.

c. Safety relief valve operation may result in direct pressurization of primary containment.

d. The unsafe region of the Heat Capacity Level Limit (HCLL) curve will be entered

97 **QUESTION:**

Given the following conditions:

- The plant is shutdown
- The reactor vessel head is installed
- No core alterations are planned
- A loss of all normal shutdown cooling has occurred.

Which of the following is an APPROVED alternate method of decay heat removal for the above conditions?

a. Filling the main steam lines with the condensate pumps and discharging to the suppression pool via the Safety Relief Valves (SRV). The suppression pool is cooled by RHR.

b. Gravity draining reactor coolant through the RHR heat exchangers to the suppression pool. The cooled water will be returned to the vessel by LPCS.

c. Circulating reactor coolant from the RPV to the main condenser via the MSIVs and MSL drains. The cooled water will be returned to the vessel by the condensate system.

d. Start one reactor recirculation pump and place RHR in the Fuel Pool Cooling Assist Mode.

98 QUESTION

Given the following conditions:

- A loss of coolant accident is in progress
- An Auxiliary Building entry into the HPCS Pump room is needed
- Radiation levels in that area do NOT allow personnel access

Which of the following describe the radiation levels in the HPCS Pump room area?

HPCS Pump room radiation levels are:

a. above the maximum normal operating value

b. above the maximum safe operating value.

c. greater than the level necessitating a Site Area Emergency.

d. greater than the levels required for an Emergency Depressurization.

99 QUESTION:

Given the following conditions:

- The plant was operating at 55% power
- A high drywell pressure condition has occurred due to a leak
- All expected automatic actions have occurred

Which of the following will be the MAXIMUM expected annulus pressure for these conditions?

a. -0.75 inches W.G.

b. -0.50 inches W.G.

c. -0.25 inches W.G.

d. 0.00 inches W.G.

100 QUESTION:

Given the following conditions:

- A failure to scram has occurred
- Power is 20% with control rods being inserted normally
- EOP-3, "Secondary Containment Control" has been entered due to HVAC cooler high differential temperatures caused by a fire in the Auxiliary Building
- The Main Steam Isolation Valves have closed
- Reactor water level control is via Condensate/Feedwater

Which of the following systems should be isolated if it is discharging into the Auxiliary Building?

a. Control Rod Drive Hydraulics

b. Reactor Water Cleanup

c. Fire suppression systems

d. Feedwater.