ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-458/95-13

License: NPF-47

Licensee: Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana

Facility Name: River Bend Station

Inspection At: St. Francisville, Louisiana

Inspection Conducted: August 14 through September 1, 1995

Inspectors: H. F. Bundy, Chief Examiner, Operations Branch
T. R. Meadows, Examiner, Operations Branch

Accompanying Personnel: M. Parrish, Examiner, Lockheed Idaho Technologies Company
S. Willoughby, Examiner, Lockheed Idaho Technologies Company
F. S. Jaggar, Program Manager, Lockheed Idaho Technologies Company

Approved: Joseph I. Tapia, Acting Chief, Operations Branch, Division of Reactor Safety — 9/26/95 Date

Inspection Summary

Areas Inspected: Routine, announced inspection of the qualifications of applicants for operator licenses at the River Bend Station facility, which included an eligibility determination and administration of comprehensive written and operating examinations. The examination team also observed the performance of onshift operators and plant conditions incident to the conduct of the in-plant applicant evaluations. The examiners used the guidance provided in NUREG-1021, "Operator Licensing Examiner Standards," Revision 7, Supplement 1, Sections 201-204, 301-303, 401-403, issued June 1994.

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Results:

Plant Operations

- Four of five applicants for reactor operator licenses and five of six applicants for senior reactor operator licenses satisfied the requirements of 10 CFR 55.33(a)(2) and have been issued the appropriate licenses (Section 1).

- The lesson plans supplied by the licensee for examination development were clearly written and understandable. However, the licensee's post-examination review revealed a few discrepancies between the lesson plans and other facility documents used by operations (Section 1.1).

- The minimum passing score for the written examination was 80 percent. One reactor operator applicant failed the written examination with a score of 77.3 percent, while passing scores ranged from 83.5 to 86.6 percent. One senior reactor operator failed the written examination with a score of 76.3 percent, while passing scores ranged from 80.4 to 91.8 percent (Section 1.2).

- All applicants performed well on the operating test part of the examination and there were no failures (Section 1.3).

- Decorum of control room operators and other plant personnel was appropriate (Section 1.4).

- A plant procedure for the fire protection water system was not appropriately human factored (Section 1.4).

Plant Support

- Housekeeping and plant appearance were excellent (Section 1.4).

Summary of Inspection Findings:

- No items requiring a tracking number were identified in this inspection.

Attachments:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Simulation Facility Report
- Attachment 3 - Written Examination and Answer Key
- Attachment 4 - Facility Licensee Post Examination Comments

DETAILS

1 LICENSED OPERATOR APPLICANT INITIAL QUALIFICATION EVALUATION (NUREG-1021)

During the inspection, the examiners evaluated the qualifications of 11 applicants: five applicants for reactor operator (RO) licenses, five applicants for senior reactor operator (SRO) licenses, and one applicant for upgrade to SRO from a current RO license. The inspection assessed the eligibility and administrative and technical competency of the applicants to be issued licenses to operate and direct the operation of the reactivity controls of a commercial nuclear power facility in accordance with 10 CFR Part 55 and NUREG-1021, "Operator License Examiner Standards," Revision 7, Sections 200 (series), 300 (series), and 400 (series). Furthermore, the inspection included evaluations of facility materials, procedures, and simulation capability used to support development and administration of the examinations. These areas were evaluated using the guidance provided in the areas of NUREG-1021 cited above. Additionally, the examination team also observed the performance of on-shift operators and plant conditions incident to the conduct of in-plant applicant evaluations.

After completion of the evaluations, the examiners recommended that four applicants for RO and five applicants for SRO licenses satisfied the requirements of 10 CFR 55.33(a)(2).

1.1 Facility Materials Submitted for Examination Development

The examiners reviewed the licensee's materials provided for development of the examination, which included station administrative and operating procedures, lesson plans, question banks, simulator scenarios, and job performance measures. The materials presented were adequate in scope, depth, and variety, and were used extensively in development of examination materials. The lesson plans supplied were clearly written and understandable. However, the licensee's post-examination review revealed a few discrepancies between the lesson plans and other facility documents used by operations as discussed in Attachment 4. As discussed below, the written examination answer key was revised to reflect the best available information.

1.2 Written Examinations

The examiners developed comprehensive RO and SRO written examinations in accordance with the guidelines of NUREG-1021, Revision 7, Supplement 1, Section 401. The examinations consisted of 100 multiple choice questions. During the week of July 31, 1995, members of the facility operations and training departments, under the nondisclosure security provisions of NUREG-1021, reviewed the examinations. The examiners incorporated the facility preadministration review comments and administered the examinations to the license applicants on August 18, 1995.

The chief examiner provided the facility training staff with a copy of the as administered written examinations together with answer keys and their pre-

The chief examiner provided the facility training staff with a copy of the as administered written examinations together with answer keys and their pre-examination review comments immediately following the completion of the examination by the applicants. The licensee submitted comments on four questions (Attachment 4), which were common to both the RO and SRO examinations. The chief examiner analyzed the comments, using the supporting information supplied by the licensee, and determined that the facility comments were technically correct and the actions requested by the facility were in accord with NUREG-1021, Revision 7, Supplement 1. Therefore, two correct answers were allowed for two questions RO 24/SRO 27 and RO 47/SRO 46 and two questions were deleted RO 52/SRO 49 and RO 75/SRO 71.

After incorporating the above accepted comments from the licensee to the answer key, the examiners reviewed applicant performance on individual questions and observed that the following questions were missed by 50 percent or more of the applicants attempting to answer the question. The questions are referenced here by exam type and question number. Refer to Attachment 3 for the specific questions and answers.

Analyzed questions: SRO 1(RO 2), RO 1, SRO 6(RO 6), SRO 17, SRO 52(RO 57), SRO 54(RO 60), RO 64, SRO 73(RO 77), SRO 87, SRO 96(RO 96), and SRO 98

Further question analysis by the chief examiner resulted in further changes to the answer key as follows:

- Question SRO 1(RO 2) related to Overtime Guidelines for the At-The-Controls Operator. It was determined that the answer should have been (a) instead of (d) and the answer key was changed to reflect this for the SRO examination. After the correction, only one SRO applicant missed this question. Subsequent discussions with licensee representatives revealed that ROs are not held accountable for knowing the specific guidelines from memory. The overtime assignments are made by supervisory personnel. Therefore, Question RO 2 was deleted.

- Question SRO 6(RO 6) related to the use of human tags for equipment clearances. Choice (c) was given in the answer key. However, many of the applicants selected Choice (b) which stated that the duration of the evolution must be less than one hour. Because the administrative procedure says that the human tag may be used for maintenance and testing of short duration, and one hour is short duration, Choice (b) is also considered correct and the answer key was changed to reflect this for both examinations.

- Question SRO 98 related to the specific leak rate expected due to failure of a upper recirculation pump seal. Four of the six applicants selected Choice (b), 5.0 gpm, as the answer, whereas Choice (a), 0.5 gpm, was the correct answer. The staff determined that what is expected from a safe operator is to know that a moderate leak to the drywell will occur as a result of the seal failure and be able to address it in accordance with Technical Specifications and plant procedures. The same actions would be taken for either leak rate. Therefore, this question was deleted because it is not discriminatory.

Initially both examinations consisted of 100 questions valued at one point each for a total of 100 points. After deleting the questions discussed above both examinations consisted of 97 questions. Two answers were accepted for three questions on each examination (6, 27, and 46 on the SRO examination and 6, 24, and 47 on the RO examination).

Overall, applicants performed satisfactorily on the written examinations. Four of the five RO applicants passed with scores ranging from 83.5 to 86.6 percent. One RO applicant failed with a score of 77.3 percent. Five of the six SRO applicants passed with scores ranging from 80.4 to 91.8 percent. One SRO applicant failed with a score of 76.3 percent. The Chief Examiner concluded that no specific area of significant knowledge weakness was apparent in response to the examination questions. Therefore, this information is provided to the facility training staff for consideration as feedback into future training needs.

1.3 Operating Tests

The examiners developed comprehensive operating tests in accordance with the guidelines of NUREG-1021, Revision 7, Supplement 1, Section 301. The operating tests consisted of three parts: administrative topics, control room systems and facility walkthrough, and integrated plant operations. The examiners previewed and validated the various parts of the operating tests at the River Bend Station site during the week of August 14, 1995, with the assistance of facility training and operations personnel under security agreement. The examiners administered the operating tests during the week of August 28, 1995. Overall, the applicants performed well on the operating test part of the examination in that crew communications were generally good and competency levels were generally high. There were no failures on the operating test part of the examination.

1.3.1 Integrated Plant Operations

The examiners evaluated three crews using the River Bend Station plant-specific simulation facility. It was originally intended to evaluate the first crew on four scenarios with each candidate (two RO and two SRO applicants) participating in three of the four scenarios. However, it was necessary to modify the crew composition for the fourth scenario to evaluate an SRO applicant performing in an RO position during a major transient. This resulted in one RO applicant participating in only two scenarios and one SRO applicant (with his concurrence) participating in all four scenarios. The second crew (one initial SRO, the upgrade SRO, and two RO applicants) were evaluated on three scenarios with the upgrade SRO and two RO applicants each participating in two scenarios and the initial SRO applicant participating in all three scenarios. The third crew (one RO and two SRO applicants) were evaluated on three scenarios with each candidate participating in each scenario.

The examination team noted good communication practices among crew members. The applicants generally practiced echo and confirmation communication prior to acting. In instances where an applicant failed to follow this practice, the other party to the communication followed up to ensure that the communication was understood. Crew briefings by the SRO applicants were generally effective and timely. No generic weaknesses were noted during the conduct of the dynamic simulator examinations.

All eleven applicants passed this part of the operating test.

1.3.2 Administrative Topics and Control Room Systems/Facility Walk-Through

The examiners evaluated the administrative capability and system operations ability of each of the applicants, using job performance measures and prescripted questions. The tasks and questions related to the scope of potential duties of a licensed reactor operator. The administrative part used job performance measures or prescripted questions to assess the ability of the applicants to carry out their various administrative responsibilities. The scope of system-related tasks included nonlicensed operator tasks outside the control room. The applicants performed some of the tasks in the simulation facility in the dynamic mode. They simulated the remainder of the tasks in the plant control room and at local operating stations throughout the plant through discussions. To further assess system knowledge, the examiners asked prescripted questions relating to the system involved in each task. The questions solicited short-answer responses and permitted the applicants to use operationally controlled references to aid in their responses when appropriate.

Each applicant was required to enter the protected, vital, and radiation control areas to complete one or more tasks. Applicants were familiar with facility escort procedures on entering these areas with a visiting examiner. The applicants properly followed facility radiation control and foreign material exclusion procedures. The examiners noted that applicants were observant of activities in the plant.

The examiners combined the applicant's task performance and followup questions in accordance with the guidelines of NUREG-1021, Revision 7, Supplement 1, Section 303, to evaluate performance on these parts of the operating test. All applicants passed these parts of the operating test. The candidates' performances were generally good and no generic weaknesses were observed.

1.4 Observations

The examination team observed the performance of on-shift operators and plant conditions incident to the conduct of in-plant applicant evaluations. These observations did not impact the evaluation of individual applicants.

Housekeeping and plant appearance were excellent. Decorum of control room operators and other plant personnel was appropriate.

During the facility walk-through part of the operating test, it was noted that a plant procedure was not appropriately human factored. Step 4.2.3.1 in Procedure SOP-0037, "Fire Protection Water System Operating Procedure," Revision 11B, required opening the fuel supply solenoid by turning the knurled manual knob. While administering the operating test, the examiners noted that there were two unlabeled knurled manual knobs on the fuel supply solenoid. The procedure did not contain a sketch to indicate the correct knob. A licensee representative stated that they would resolve this specific issue.

1.5 Simulation Facility

During preparation and conduct of the operating tests, the examination team did not observe any simulator fidelity discrepancies which affected exam administration. However, modeling limitations, as described in Attachment 2, were observed. During performance of the job performance measures, applicants identified labeling errors involving the reactor protection system white status lights. It was subsequently determined that Station Problem Report 1536 had been issued in November 1994 to correct this configuration control problem. These lights were labeled correctly in the plant control room.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

J. Baggett, Supervisor, Initial Operations Training
J. Fralick, Operations Instructor
A. Johnson, Operations Instructor
M. Krupa, Manager, Operations
J. McGaha, Vice President, Operations
J. Summers, Licensing Specialist
W. Trudell, Operations Superintendent
L. Woods, Operations Shift Superintendent

1.2 NRC Personnel

W. Smith, Senior Resident Inspector

In addition to the personnel listed above, the examiners contacted other personnel during this inspection period.

All personnel listed above attended the exit meeting.

2 EXIT MEETING

An exit meeting was conducted on September 1, 1995. During this meeting, the chief examiner reviewed the scope and generic findings of the inspection. The chief examiner did not disclose preliminary results of individual evaluations since they are subject to change during the final review and approval process. The licensee did not identify, as proprietary, any information provided to, or reviewed by, the examiner. The licensee did not state any position on the findings presented during the exit meeting.

ATTACHMENT 2

SIMULATION FACILITY REPORT

Facility Licensee: River Bend Station

Facility Docket: 50-458

Operating Tests Administered at: River Bend Station

Operating Tests Administered on: August 28 to September 1, 1995

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.

- The reactor protection system white status lights on the main control board were incorrectly labeled.

- A limited number of simulator instructor addressable instrument malfunctions were available. Those available largely addressed nuclear instrumentation failures. For example, a high reactor pressure half scram could not be modeled.

<u>ATTACHMENT 3</u>

WRITTEN EXAMINATION AND ANSWER KEY

SRO

Nuclear Regulatory Commission
Operator Licensing
Examination

This document is removed from
Official Use Only category on
date of examination.

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 4

CANDIDATE'S NAME: ______________________

FACILITY: River Bend 1

REACTOR TYPE: BWR-GE6

DATE ADMINISTERED: 95/08/18

INSTRUCTIONS TO CANDIDATE:

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

TEST VALUE	CANDIDATE'S SCORE	%	
98.00 ~~100.00~~	______ FINAL GRADE	___ %	TOTALS

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

____
Candidate's Signature

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

MULTIPLE CHOICE

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ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

046	a	b	c	d	___	069	a	b	c	d	___
047	a	b	c	d	___	070	a	b	c	d	___
048	a	b	c	d	___	071	a	b	c	d	___
049	a	b	c	d	___	072	a	b	c	d	___
050	a	b	c	d	___	073	a	b	c	d	___
051	a	b	c	d	___	074	a	b	c	d	___
052	a	b	c	d	___	075	a	b	c	d	___
053	a	b	c	d	___	076	a	b	c	d	___
054	a	b	c	d	___	077	a	b	c	d	___
055	a	b	c	d	___	078	a	b	c	d	___
056	a	b	c	d	___	079	a	b	c	d	___
057	a	b	c	d	___	080	a	b	c	d	___
058	a	b	c	d	___	081	a	b	c	d	___
059	a	b	c	d	___	082	a	b	c	d	___
060	a	b	c	d	___	083	a	b	c	d	___
061	a	b	c	d	___	084	a	b	c	d	___
062	a	b	c	d	___	085	a	b	c	d	___
063	a	b	c	d	___	086	a	b	c	d	___
064	a	b	c	d	___	087	a	b	c	d	___
065	a	b	c	d	___	088	a	b	c	d	___
066	a	b	c	d	___	089	a	b	c	d	___
067	a	b	c	d	___	090	a	b	c	d	___
068	a	b	c	d	___	091	a	b	c	d	___

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

092	a	b	c	d	___
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099	a	b	c	d	___
100	a	b	c	d	___

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

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must 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. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

8. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

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

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. Proportional grading will be applied. Any additional wrong information that is provided may count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

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

14. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

15. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

16. To pass the examination, you must achieve a grade of 80% or greater.

17. There is a time limit of four (4) hours for completion of the examination.

18. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

A licensed operator has been assigned as the At-The-Controls Operator for the following periods:

-- Friday	1600 to 0400
-- Saturday	1200 to 2400
-- Sunday	0800 to 1600
-- Monday	0800 to 1600
-- Tuesday	0800 to 2400
-- Wednesday	0800 to 2000

(Exclude turnover time)

On which day did he first violate the Overtime Guidelines?

a. Sunday

b. Monday

c. Tuesday

d. Wednesday

QUESTION: 002 (1.00)

With the SS, CRS, STA, ATC operator ineligible as members, the fire brigade shall consist of at least:

a. four members not including one NEO and one other person required for safe shutdown of the plant.

b. four members including one other person not required for safe shutdown of the plant.

c. five members not including one NEO nor other personnel required for safe shutdown of the plant.

d. five members including one NEO and other personnel required for safe shutdown of the plant.

QUESTION: 003 (1.00)

The Administrative Control Room Supervisor should be stationed on startups from:

a. criticality until the first bypass valve is 50% open.

b. criticality until the last FWREG Valve is placed in service.

c. withdrawal of the first control rod until criticality.

d. withdrawal of the first control rod until the last FWREG Valve is placed in service.

QUESTION: 004 (1.00)

Which of the following describes the MINIMUM permission and notification requirements that should be obtained in order to enter Technical Specification 3.0.3. for maintenance or testing?

Permission is required from:

a. the Operations Superintendent with notification to the NRC Resident Inspector.

b. the Operations Superintendent and NRC Resident Inspector with notification to the NRC Operations Center.

c. the General Manager-Operations with notification to the NRC Resident Inspector.

d. the Shift Superintendent with notification to the Manager-Operations and NRC Resident Inspector.

QUESTION: 005 (1.00)

When performing protective tagging, who has the PRINCIPLE responsibility to ensure that proper isolation has been performed AND proper tagging is installed for employee protection?

a. the individual workers.

b. the person hanging the tags.

c. the person in charge of the work.

d. the person approving the clearance.

QUESTION: 006 (1.00)

Which of the following is a requirement for using a Human Tag?

a. V. P. Nuclear approval must be obtained.

b. Duration of the evolution must be less than one hour.

c. The supervisor of the job is responsible for ensuring that no one operates the device.

d. No more than three isolation points may be held by Human Tags at one time for a given maintenance activity.

QUESTION: 007 (1.00)

When tagging a component out of service, the general sequence should be:

a. Control switch, breaker, discharge valve, suction valve, vents and drains.

b. Breaker, control switch, suction valve, discharge valve, vents and drains.

c. Breaker, control switch, discharge valve, suction valve, vents and drains.

d. Control switch, breaker, suction valve, discharge valve, vents and drains.

QUESTION: 008 (1.00)

The Shift Supe intendent is responsible for evaluating each condition report to determine:

a. The severity level.

b. The Q class.

c. Twenty-four hour reportability.

d. Immediate reportability.

QUESTION: 009 (1.00)

A Health Physics Technician has labeled a check valve as a "Hot Spot".

The MINIMUM contact dose rate that an operator should expect from this check valve is greater than:

a. 50 mRem

b. 100 mRem

c. 200 mRem

d. 500 mRem

QUESTION: 010 (1.00)

A declared pregnant non-licensed operator is assigned to your crew for training. During the duration of her pregnancy her exposure is administratively limited to:

a. 400 mRem and she is not allowed to enter high radiation areas.

b. 500 mRem and she is not allowed to enter any radiation areas.

c. 400 mRem and she is not allowed to enter areas posted for neutron radiation.

d. 500 mRem and she is not allowed to enter areas posted for noble gases.

QUESTION: 011 (1.00)

Which of the following requirements for drywell entry may be waived solely by the Shift Superintendent?

a. Drywell air sample must be taken upon entry.

b. Drywell must be vented for at least one hour prior to entry.

c. Standby individuals must be provided for the entry.

d. Drywell hydrogen concentration must be less than 4%.

QUESTION: 012 (1.00)

You are performing an in-plant surveillance test and leave the work area during the surveillance for a break.

In order to restart the surveillance test, in addition to notifying he OSS/CRS, you are required to verify at a MINIMUM that:

a. Only the prerequisites and all precautions and limitations are met.

b. Only that previously performed steps are in the same state.

c. Only that the prerequisites are still met.

d. Only that the prerequisites are met and previously performed steps are in the same state.

QUESTION: 013 (1.00)

You can verify that an operator aid has been approved if it contains:

a. the Department Foreman's signature.

b. the signature of an OSS or CRS.

c. the equipment lable request number.

d. a work document number.

QUESTION: 014 (1.00)

In accordance with River Bend Station administrative requirements, deviations to procedures which are taken to protect the health and safety of the public must be approved by:

a. the Manager-Operations.

b. any SRO with an active license.

c. the Shift Superintendent or Control Room Supervisor.

d. the Operations Superintendent with notification to the NRC Resident Inspector.

QUESTION: 015 (1.00)

Given the following conditions:

-- The plant is in cold shutdown.
-- Shutdown cooling is in service.

Which of the following actions would require review and approval from senior plant management?

a. Opening the breaker for 1E12-F008 RHR Shutdown Cooling Outboard Isolation Valve with the valve open.

b. Opening the breaker for 1E12-F009 RHR Shutdown Cooling Inboard Isolation Valve with the valve open.

c. Lifting leads such that 1E12-F024A RHR Pump A Test Return to Suppression Pool can be opened with 1E12-F006A RHR Pump A SDC Suction Valve already open.

d. Lifting leads such that 1E12-F006A RHR Pump A SDC Suction Valve can be opened when 1E12-F004A RHR Pump A Suppression Pool Suction Valve is open.

QUESTION: 016 (1.00)

During the implementation of the emergency plan, you have a need for a direct computer link with off-site agencies.

Which of the following off-site agencies DO NOT have a direct computer link with River Bend Station?

a. Nuclear Regulatory Commission

b. State of Louisiana

c. Parishes within the 10-mile EPZ

d. State of Mississippi

QUESTION: 017 (1.00)

Which of the following would be an UNEXPECTED result of a post-accident hydrogen deflagration?

a. Over-pressurization of the containment.

b. Uncovering the horizontal vents.

c. Structural damage to containment.

d. Failure of equipment in containment which may be useful in a post burn environment.

QUESTION: 018 (1.00)

In order to provide shielding, the TIP detectors are normally stored:

a. inside the reactor vessel and above the core.

b. inside the reactor vessel and below the core.

c. within the "in-shield" position inside the TIP room.

d. within the indexing mechanism.

QUESTION: 019 (1.00)

If all normal fill systems are unavailable, which of the following systems would first be used to fill the upper pool?

a. Standby Service Water

b. Normal Service Water

c. Circulating Water

d. Turbine Plant Component Cooling Water

QUESTION: 020 (1.00)

Which of the following describes a properly oriented fuel bundle?

a. The orientation boss on the fuel assembly handle points away from the control rod.

b. The channel spacer buttons are adjacent to the control rod and adjacent to each other.

c. Serial number on the handle is readable from the outside edge of the four bundle fuel cell.

d. The channel fasteners are located on the outside edge of the four bundle fuel cell.

QUESTION: 021 (1.00)

Given the following conditions:

-- The plant is operating at 75% power.
-- The Control Room Operator places the Outboard MSIV Positive Leakage Control System switch to OPERATE.
-- Special risk based surveillances are being systematically performed on all safety related instrumentation systems.

Which of the following describes why the Outboard MSIV Positive Leakage Control System will NOT initiate?

a. A LOCA signal on either high drywell pressure or low reactor water level is not present due to a surveillance.

b. The post LOCA 20 minute timer has not timed out due to a surveillance.

c. The required main steam line pressure and reactor pressure requirements have not been met.

d. All Main Steam Isolation Valves have not been fully closed.

QUESTION: 022 (1.00)

The 1CNM-P1A Condensate Pump was started at 1045 and immediately tripped. The pump was restarted at 1110 and tripped at 1130.

The soonest this condensate pump may be restarted is:

a. immediately.

b. at 1145.

c. at 1210.

d. only when the motor cools to ambient.

QUESTION: 023 (1.00)

Five minutes after a LOCA the following conditions exist:

-- Reactor water level is -90 inches and decreasing.
-- Drywell pressure is 2.2 psig and increasing.

Which of the following describes the status of Division I and Division II safety-related loads.

a. Only ECCS motor loads are sequenced onto the safety buses. Existing vital loads on the buses continue to operate.

b. All loads on the safety buses are shed and vital loads are then sequenced onto the buses.

c. Only ECCS motor loads are sequenced onto the safety buses. Existing vital loads are shed.

d. ECCS loads are not yet sequenced onto the safety buses. Existing loads on the buses continue to operate.

QUESTION: 024 (1.00)

Normal Station Transformer 1STX-XNS1C should not be used to supply:

a. 1E22*S004 (HPCS standby bus) because there is no sync check permissive in the fast transfer from normal to preferred source.

b. 1ENS*SWG1A (standby bus) because there is no sync check permissive in the fast transfer from normal to preferred source.

c. 1E22*S004 (HPCS standby bus) to preclude the possibility of heating the transformer beyond the cooling fan capacity.

d. 1ENS*SWG1B (standby bus) to preclude the possibility of heating the transformer beyond the cooling fan capacity.

QUESTION: 025 (1.00)

Unless the backup charger is connected to the HPCS bus, the Backup Charger Supply Breaker (1E22-CB10) from 1E22-PNLS001 should be locked in the open position.

This ensures that:

a. the backup battery charger is not overloaded.

b. the backup charger continuously meets Technical Specification operability requirements.

c. separation between redundant Safety-Related systems is maintained.

d. damage to the rectifier stack does not occur.

QUESTION: 026 (1.00)

Given the following conditions:

- -- Reactor water level is -90 inches and decreasing.
- -- Drywell pressure is 2.2 psig and increasing.
- -- An outside fire has caused smoke in the Control Room.
- -- The operator has attempted to manually place the Control Room ventilation in the purge mode.

Under these conditions the Control Room Smoke Removal Damper (AOD 107/108) will:

a. open and the Smoke Removal Fan will start.

b. open but the Smoke Removal Fan will be interlocked off.

c. remain closed and the Smoke Removal Fan will run on recirc.

d. remain closed and the Smoke Removal Fan will be interlocked off.

QUESTION: 027 (1.00)

If total feedwater flow drops below the RRS interlock level, the Reactor Recirculation pumps will downshift to slow speed.

What is the PRIMARY reason for this interlock?

a. pump cavitation.

b. flow control valve cavitation.

c. excessive axial thrust on the pump.

d. inaccurate wide range level indication.

QUESTION: 028 (1.00)

With both Recirculation loops in operation and core flow at 62% of rated, Recirculation loop flow mismatch must be maintained within:

a. 5% of rated Recirculation flow in order to prevent excessive jet pump vibration.

b. 10% of rated Recirculation flow in order to prevent excessive jet pump vibration.

c. 5% of rated Recirculation flow in order to prevent a non-conservative APRM flow bias signal.

d. 10% of rated Recirculation flow in order to prevent a non-conservative APRM flow bias signal.

QUESTION: 029 (1.00)

The following conditions exist:

-- The plant was operating at 100% power.
-- A loss of BOTH Reactor Protection System (RPS) buses has occurred

Which of the following describes the operators responsibilities regarding the Main Steam Isolation Valves (MSIV) for this failure?

The MSIV:

a. control switches must be placed in "Close" before the valves begin to drift closed.

b. low vacuum isolation must be bypassed before reenergizing the RPS buses.

c. control switches must be placed in "Close" to reset the MSIV isolation, once power is restored to RPS.

d. control switches must be placed in "Close" before reenergizing the RPS buses.

QUESTION: 030 (1.00)

The HPCS Suppression Pool Suction Valve (F015) will automatically open:

a. when the HPCS CST Suction valve (F001) automatically goes closed with a HPCS initiation signal present.

b. anytime a low CST level or high suppression pool level exists.

c. only when a HPCS initiation signal is present AND either a low CST level OR high suppression pool level exists.

d. two seconds after the HPCS CST Suction valve (F001) automatically goes closed.

QUESTION: 031 (1.00)

The following conditions exist:

-- The plant has experienced a loss of all AC power (BLACKOUT)
-- The Reactor Core Isolation Cooling (RCIC) system is running and injecting to the reactor.

RCIC then receives a valid Div.1 and Div.2 isolation signal.

Which of the following valves will CLOSE under these conditions?

Note: Only consider direct actions from a system isolation signal.

a. Turbine steam supply warmup valve-- F076

b. Injection shutoff valve-- F013

c. Turbine steam supply inboard isolation valve -- F063

d. Pump suppression pool suction valve -- F031

QUESTION: 032 (1.00)

Given the following conditions:

- -- The Reactor Water Cleanup (RWCU) system is operating in the normal mode.
- -- The RWCU Keylock Bypass Switches (E31-S1A,B) on P632 and P642 have been placed in "Bypass".

Select the expected effect on the RWCU system.

a. The RWCU system isolation on initiation of the Standby Liquid Control System is defeated.

b. The RWCU system isolation from the Leak Detection System is defeated.

c. All RWCU isolations, except for high differential flow and high area temperature are defeated.

d. All RWCU system isolation signals are defeated.

QUESTION: 033 (1.00)

Which of the following plant systems are required to be operable for SECONDARY Containment Integrity?

a. Combustible Gas Control System.

b. MSIV Positive Leakage Control System.

c. Penetration Valve Leakage Control System.

d. Standby Gas Treatment System.

QUESTION: 034 (1.00)

Which of the following will start the Division I Standby Diesel Generator but will NOT result in closure of the generator output breaker.

(Assume normal ENS Bus voltage is exactly 4160 VAC.)

a. ENS bus voltage has been 3300 VAC for 5 seconds.

b. Reactor water level is steady at -135 inches.

c. The "Arm and Depress" pushbutton for Low Pressure Core Spray/RHR has been depressed.

d. Drywell pressure is steadily increasing and has reached 1.50 psig.

QUESTION: 035 (1.00)

The following conditions exist:

-- The Div 2 standby diesel generator is loaded and in parallel with bus 1ENS*SWG1B through the normal breaker.
-- A LOCA signal occurs.

Which of the following describes the effect of the standby diesel generator and bus 1ENS*SWG1B?

a. The normal breaker to the bus will open and the diesel generator will supply bus loads.

b. The normal breaker and diesel output breaker will open then after loads are stripped, the diesel breaker will reclose .

c. The diesel generator output breaker will open and cannot be closed until the LOCA is reset.

d. The diesel generator output breaker will remain closed in parallel operation with the bus.

QUESTION: 036 (1.00)

Which of the following will result in an automatic Control Room ventilation system isolation?

a. Two or more control room smoke alarms.

b. Any automatic start of the Standby Gas Treatment system.

c. Any control building area radiation monitor alarm.

d. Control building local intake high radiation signal.

QUESTION: 037 (1.00)

The following conditions exist:

-- The reactor is at 100% power.
-- The Off-Gas Post Treatment HI-HI-HI radiation alarm (P601/22A/A03) has occurred.
-- The Offgas System automatically isolated, (1N64-F060 Off-Gas Discharge to Vent valve is closed).

Which of the following actions is PROHIBITED?

a. Purge the Off-Gas system with service air.

b. Shift to the Standby Off-Gas Component.

c. Place steam seals on auxiliary steam.

d. Reduce power below 60%.

QUESTION: 038 (1.00)

The following conditions exist:

- -- The plant is operating at 100% power.
- -- An operator initiated scram becomes necessary.
- -- The mode switch is placed in "Shutdown" and then Manual Scram pushbuttons are depressed.
- -- All plant systems respond as designed.

Considering RPS logic, select the ACTUAL scram signal that caused the Reactor Protection System to deenergize the scram solenoids.

a. Manual Scram pushbuttons depressed direct to scram solenoids.

b. Control Valve fast closure trip.

c. APRM Neutron Flux - High, Setdown logic train.

d. MSIV closure.

QUESTION: 039 (1.00)

A startup is in progress and the following conditions exist:

-- Reactor power	- 2% on APRMs
-- Reactor water level	- 35 inches
-- Reactor pressure	- 550 psig
-- Condenser vacuum	- 20 inches
-- Mode switch	- in STARTUP
-- MSIVs	- open

Which of the following describes the effect if RPS bus "A" is deenergized?

a. All MSIVs will close.

b. MSIVs "A" and "C" will close.

c. A Div 1 half scram will occur.

d. A full scram will occur.

QUESTION: 040 (1.00)

Which of the following RPS trip signals is never bypassed?

a. APRM Neutron Flux - High, Setdown.

b. IRM Neutron Flux - High.

c. Reactor low water level.

d. Mode Switch in SHUTDOWN.

QUESTION: 041 (1.00)

During a scram from 100% power, which of the following describes the response of a withdrawn control rod if the accumulator piston for its hydraulic control unit does NOT move?

NOTE: Assume the accumulator is mechanically bound up.

The control rod will:

a. not insert.

b. insert at slower than normal speed.

c. insert after the CRD flow control valve opens.

d. insert only if CRD charging pressure is greater than 400 psig.

QUESTION: 042 (1.00)

During a startup conducted within 24 hours of previous high power operation, unusually high control rod worths may be experienced.

The range of control rod positions where this condition is particularly likely to occur are notch positions:

a. 0 - 16

b. 8 - 20

c. 12 - 24

d. 16 - 32

QUESTION: 043 (1.00)

While at 100% power, which of the following describes the effect on control rods of losing power to Alternate Rod Insertion (ARI) channel A AND channel B?

ARI will:

a. not initiate either automatically or manually from the control room.

b. not initiate automatically but will initiate manually from the control room.

c. isolate the air supply but not vent the scram air header.

d. automatically initiate.

QUESTION: 044 (1.00)

Which of the following describes a requirement for conducting control rod coupling checks?

a. For all control rods, prior to reactor criticality after completing core alterations.

b. Anytime the control rod is withdrawn to the full out position.

c. Only the first time that each control rod reaches position 48 after a reactor startup.

d. The first time that each control rod reaches position 48 after a reactor startup and weekly thereafter.

QUESTION: 045 (1.00)

The APRM Channel A meter function switch is placed in the "COUNT" position.

Which of the following is the MINIMUM indication for the APRM to be considered operable independent of level requirements?

a. 45%

b. 50%

c. 55%

d. 60%

QUESTION: 046 (1.00)

Which of the following will prevent RCIC discharge to the CST through the test line return valves (F022, F059)?

a. The CST suction valve is open (F010).

b. RCIC minimum flow valve open (F019).

c. CST level is less than 6.5 inches.

d. Reactor vessel level is 55 inches.

QUESTION: 047 (1.00)

The division 1 diesel generator is started using the local "EMERGENCY START" PUSHBUTTON:

Which of the following diesel generator shutdown signals is bypassed in this condition?

a. Generator differential current.

b. Low diesel lube oil pressure.

c. Local manual stop pushbuttons.

d. Diesel overspeed.

QUESTION: 048 (1.00)

Which of the following require loss of power to both divisions of their control logic in order to cause an automatic isolation?

a. Residual Heat Removal valves

b. Main Steam Isolation valves

c. Reactor Water Cleanup valves

d. Reactor Core Isolation Cooling valves

QUESTION: 049 (1.00)

While in a refueling outage, which of the following requires Primary Containment Integrity to be established?

a. Both trains of Standby Gas Treatment become inoperable.

b. One LPRM detector will be replaced with a new one.

c. All source range detectors are discovered to be inoperable.

d. Irradiated fuel is to be moved in the lower fuel pool.

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QUESTION: 050 (1.00)

Which of the following will bypass ALL rod blocks caused by SRM "A"?

a. All IRMs on range 8.

b. Reactor mode switch in REFUEL.

c. SRM "A" detector fully withdrawn.

d. SRM "A" function switch NOT in operate.

QUESTION: 051 (1.00)

The following conditions exist:

-- A leak inside the drywell has occurred.
-- All RHR pumps are running.
-- The Automatic Depressurization System automatically actuated at 105 seconds.
-- All ADS SRVs are open.
-- Reactor water level is now steady at - 150 inches.
-- Drywell pressure is now 1.5 psig and decreasing.
-- Reactor pressure is 200 psig.

If both Div 1 and Div 2 ADS TIMER/LEVEL 3 SEAL-IN RESET buttons are depressed and then released, which of the following describes the result on the Automatice pressurization System?

The ADS SRVs will:

a. close and then reopen after 5 minutes plus 105 seconds.

b. close and then reopen after 105 seconds.

c. close and remain closed.

d. remain open.

QUESTION: 052 (1.00)

When in Operational Condition 4 with decreasing coolant temperatures, select the MAXIMUM reactor head flange temperature that REQUIRES verification every 30 minutes?

a. 60 deg F

b. 80 deg F

c. 200 deg F

d. 300 deg F

QUESTION: 053 (1.00)

Which of the following describes the PRIMARY reason why operators are cautioned not to open RWCU Drain to Main Condenser (1G33-F046) and RWCU Drain to Rad Waste (1G33-F035), at the same time?

a. May provide an unmonitored release pathway through radwaste.

b. The heat exchangers cannot remove sufficient heat to prevent system isolation.

c. The cleanup pumps may trip on low suction pressure.

d. Condenser vacuum may be reduced.

QUESTION: 054 (1.00)

With the Refueling Platform over the core, which of the following, BY ITSELF, will initiate an RCIS rod block?

a. Mode switch in STARTUP.

b. Grapple not full down.

c. Service Platform Hoist loaded.

d. Any rod not fully inserted

QUESTION: 055 (1.00)

With irradiated fuel stored in the spent fuel pool (SFP), mechanical maintenance is preparing to move a 1400 lb. motor across the pool using the fuel building overhead crane:

Why will this move not be permitted?

a. Mechanical maintenance is NOT authorized to use the fuel building overhead crane with fuel in the SFP.

b. A 1400 lb. weight exceeds the load limit of the fuel building overhead crane.

c. No load in excess of 1200 lb. is permitted to travel over the fuel storage racks by procedure.

d. Reactor engineering support is required for any evolution involving irradiated fuel.

QUESTION: 056 (1.00)

The following conditions exist:

-- A LOCA condition has occurred based on Level 1 and Hi Drywell Pressure.
-- Water level has been recovered.
-- The "A" train of RHR was placed in suppression pool cooling with the Injection Valve, 1E12*F042A, overridden closed.

Which of the following will remove the Injection Valve, 1E12*F042A, override condition?

Note - consider each choice independently.

a. Clearing all LOCA conditions.

b. Raising reactor pressure above 487 psig.

c. Placing the injection valve control switch to CLOSE then OPEN.

d. Depressing the RHR Division I/LPCS Initiation Reset pushbutton.

QUESTION: 057 (1.00)

The following conditions exist:

-- RHR shutdown cooling loop A is in operation.
-- Reactor water level is 75 inches.

Under these conditions, which of the following has NO interlock to prevent draining the reactor vessel?

a. Suppression Pool Suction Valve 1E12*F004A.

b. Shutdown Cooling Suction Valve 1E12*F006A.

c. Test Return To Suppression Pool Valve 1E12*F024A.

d. Shutdown Cooling Outboard Isolation Valve 1E12*F008A.

QUESTION: 058 (1.00)

The following conditions exist:

- -- The plant has experienced a fuel leak and the MSIVs received a high radiation isolation signal.
- -- Neither MSIV in the "A" line closed, all others closed.
- -- B21* MOV F098A breaker tripped while attempting to close the valve.
- -- The reactor successfully scrammed.
- -- RPV level dropped to 5 inches and is now stable at 30 inches.
- -- Reactor pressure is 900 psig and slowly decreasing.
- -- There is an unisolatable steam leak in the turbine building outside the steam tunnel.
- -- Various turbine building area radiation monitors indicate increasing levels.
- -- A SITE AREA EMERGENCY has been declared based upon barrier loss.
- -- Radiation surveys are in progress and have shown that ALERT levels DO

  NOT exist yet.

Based upon the conditions given, which of the following EOPs must be entered?

a. Only EOP 1 RPV Control.

b. EOP 1 RPV Control and EOP-2 Primary Containment Control.

c. EOP 1 RPV Control and EOP 3 Secondary Containment and Radioactive Release Control.

d. EOP 2 Primary Containment Control and EOP 3 Secondary Containment and Radioactive Release Control.

QUESTION: 059 (1.00)

The following conditions exist:

-- The reactor successfully scrammed.
-- RPV level dropped to -10 inches and is now stable at 30 inches.
-- EOP 1 RPV Control was entered briefly and has been exited.

During performance of the subsequent actions of AOP-0001, Reactor Scram, RPV level control becomes erratic and level again reaches -10 inches.

Select the REQUIRED action.

a. Re-enter EOP-1 at the beginning.

b. Re-enter EOP-1 at step stating "Restore and maintain RPV water level between 9.7 and 51 inches."

c. Re-enter EOP-1 only if RPV level is not quickly restored above 9.7 inches.

d. Restore and maintain RPV water level between 9.7 and 51 inches, EOP entry is not necessary.

QUESTION: 060 (1.00)

The following conditions exist:

-- The reactor has been shutdown for one day.
-- The RWCU system is not operable.
-- Both recirculation pumps are inoperable.
-- One shutdown cooling loop is operating at maximum flow..
-- RPV water level is +80 inches.
-- RPV water temperature is 195 degrees F and stable.

Shutdown cooling loop flow is lost and cannot be immediately regained or the other loop placed in service. RPV temperatures are increasing.

Which of the following operator actions is necessary?

a. Verify primary containment integrity.

b. Raise water level to the steam lines.

c. Evacuate the containment.

d. Increase service water system flow.

QUESTION: 061 (1.00)

During execution of EOP-1 (RPV control), entry into EOP-1A ATWS is NOT necessary if:

a. the reactor is shutdown now, but could become critical later.

b. all control rods are inserted to position 02.

c. all control rods are inserted to position 00 except one.

d. all control rods are inserted to position 02 except one.

QUESTION: 062 (1.00)

Should it become necessary to lower reactor level during an ATWS condition (EOP-4A Level/Power Control), the following systems are specified for use to maintain level:

Condensate/feedwater
CRD
RCIC

Only these systems are to be used because:

a. these systems provide the cleanest source of water for injection into the reactor.

b. their point of injection ensures mixing of the cold injection water with warmer water prior to core entry.

c. these systems can operate automatically so the operator need only verify lineups when this step is reached.

d. at this point in the ATWS, reactor pressure precludes use of other systems.

QUESTION: 063 (1.00)

The following conditions exist:

-- An ATWS is in progress.
-- Reactor power is 22%.
-- Reactor water level is -10 inches.
-- Reactor pressure is 960 psig.

Which of the following will be the first to be severely challenged and is of primary importance should a full MSIV closure occur?

a. Primary containment integrity.

b. Secondary containment integrity

c. Fuel integrity.

d. RPV integrity.

QUESTION: 064 (1.00)

When EOP-4, sheet 2, Emergency RPV Depressurization, permits defeating isolation interlocks in order to rapidly depressurize without SRVs, which of the following MSIV isolation signals may be bypassed?

a. Only the RPV low level 1 signal.

b. Only the RPV Low Level 1 and low main steam line pressure signal.

c. All MSIV isolation signals except main steam line high radiation.

d. All MSIV isolation signals.

QUESTION: 065 (1.00)

The following conditions exist:

-- An SRV spuriously opened and stuck open.
-- Reactor power has been reduced to 85%.
-- EOP-2, Primary Containment Control, has been entered
-- Suppression pool cooling is in service.
-- Suppression pool temperature is increasing.
-- Efforts to close the SRV have been unsuccessful.

Which of the following is the INITIAL required action as suppression pool temperature is increasing beyond 100 deg F?

a. Scram and enter AOP-0001 if temperature reaches 105 deg F.

b. Enter EOP-1, RPV Control, and scram if temperature reaches 110 deg F.

c. Emergency depressurize if temp exceeds the Heat Capacity Temperature Limit.

d. Reduce reactor pressure as necessary to maintain pressure below the Heat Capacity Temperature Limit.

QUESTION: 066 (1.00)

While conducting refueling operations, the following has cccurred.

- You are the fuel handler.
-- A fuel bundle is dropped during withdrawal from the core.
-- The bundle is resting about 30 degrees off vertical.
-- No radiation alarms have been received, but bubbles are observed rising from the fuel bundle.

Select the REQUIRED immediate action.

a. Stop all refueling operations and immediately evacuate all persons from the containment and drywell.

b. Retrieve the bundle and place it in the nearest safe position then evacuate the fuel building, containment, and drywell.

c. Move the bundle to a safe location and await instructions from the Recovery Manager.

d. Start the Annulus Mixing and Standby Gas Treatment systems and stop further fuel movements if any radiation alarm is received.

QUESTION: 067 (1.00)

Given a Mollier diagram and assuming rising RPV temperatures and pressures:

Select the first set of plant conditions for which ALL reactor water level indicators should NOT be used.

a. RPV pressure 60 psig
Drywell/Containment temperature 200 degrees F

b. RPV pressure 90 psig
Drywell/Containment temperature 300 degrees F

c. RPV pressure 200 psig
Drywell/Containment temperature 400 degrees F

d. RPV pressure 1000 psig
Drywell/Containment temperature 180 degrees

QUESTION: 068 (1.00)

The following conditions exist:

-- The reactor is in cold shutdown and vented.
-- One reactor recirculation pump is operating.
-- One shutdown cooling loop is in operation.
-- RPV level is 35 inches.

Subsequently, all shutdown cooling and forced circulation is lost and RPV level is being raised greater than 75 inches.

Select the **MAXIMUM** vessel level allowed without flooding the steam lines?

a. 75 inches

b. 105 inches

c. 140 inches

d. 196 inches

QUESTION: 069 (1.00)

Without operator action, if the Turbine Stop Valve closure scram fails to actuate, reactor protection is provided by the:

a. main steam line isolation scram.

b. recirculation pump trip (RPT) circuit.

c. reactor vessel low level (9.7 inches).

d. nuclear system high pressure scram.

QUESTION: 070 (1.00)

The following conditions exist:

-- A loss of all AC power occurred at 6:45 am.
-- Reactor pressure was initially 1030 psig.
-- The RCIC system was manually started.
-- There is a small coolant leak into the containment
-- A cooldown was initiated at 7:00 am.

The following RPV pressures have been recorded.

TIME	RPV Pressure
7:00 am	798 psig
7:15 am	610 psig

Assuming a constant rate of temperature reduction, what is the latest time that the cooldown must be secured in crder to prevent exceeding the Technical Specification for cooldown?

Note: Answer rounded to nearest minute.

a. 07:20 am

b. 07:35 am

c. 07:50 am

d. 08:05 am

QUESTION: 071 (1.00)

With the reactor initially at 100% power, a loss of instrument air to which of the following will NOT eventually result in an automatic REACTOR PROTECTION SYSTEM SCRAM?

a. Condensate and heater drain pump recirc valves.

b. Feedwater regulating valves.

c. Turbine steam seals and SJAE.

d. Scram inlet and outlet valves.

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QUESTION: 072 (1.00)

A Safety Relief Valve tailpiece vacuum breaker has failed in the open position during SRV operation.

Which of the following will result?

a. Direct pressurization of the containment air space each time the SRV is opened.

b. Steam bypassing the quenchers with a direct discharge path into the suppression pool.

c. An increase in drywell to containment differential pressure.

d. Suppression pool water being drawn up into the SRV discharge line after the SRV is closed.

QUESTION: 073 (1.00)

Which of following conditions will ALWAYS require entry into EOP-3 Secondary Containment and Radioactive release control?

a. HPCS equipment area temperature high.

b. Turbine building ventilation automatic shutdown.

c. Auxiliary Building floor drain sump overflowing.

d. SBGT system automatically started.

QUESTION: 074 (1.00)

Which of the following must be manually isolated if a high radiation condition exists in the repective area?

a. Division I(II) Annulus Ventilation.

b. Main steam line tunnel exhaust.

c. Division I(II) Fuel building ventilation exhaust.

d. Auxiliary building ventilation.

QUESTION: 075 (1.00)

Intentional entry into the Region A or B of the Power/Flow Map during normal operation is:

a. permitted if Reactor Core instability is not indicated.

b. permitted if followed by an immediate manual scram.

c. permitted if the requirements of REP-0023 are followed.

d. never permitted.

QUESTION: 076 (1.00)

With both reactor recirculation pumps initially operating in fast speed and the rod line at 101%, the following occurs:

-- The "A" recirculation pump unexpectedly trips.
-- No unstable neutron instrumentation is observed.
-- The unit is determined to be in Region B of the Power/Flow Map.

Which of the following is required ?

a. Close the "A" recirculation pump flow control valve.

b. Insert control rods in the reverse order of the control rod insertion sequence of AOP-0001 until out of Region B only .

c. Insert control rods in the reverse order of the rod sequence package until out of Region B and Region C.

d. Manually scram the reactor.

QUESTION: 077 (1.00)

What is the PRIMARY reason that EOP-2, Primary Containment Control, require emergency depressurization if you cannot restore and maintain the suppression pool level below the SRV Tail Pipe Level Limit, (21ft 3in)?

a. The capacity of the horizontal vents may be exceeded.

b. The SRV discharge lines may fail allowing steam into containment.

c. The suppression pool structural support limits will be exceeded.

d. The pressure suppression feature of the quenchers cannot be assured.

QUESTION: 078 (1.00)

The following conditions exist:

-- The reactor is critical with the mode switch in STARTUP.
-- The running Control Rod Drive (CRD) pump has tripped and cannot be immediately restarted.
-- The other CRD pump is out for maintenance.
-- The accumulator trouble alarm for the rod being moved has just been received.

A reactor scram is required:

a. immediately.

b. if any other accumulator trouble alarm occurs.

c. only if an adjacent accumulator trouble alarm occurs

d. if any valid control rod temperature alarm is received.

QUESTION: 079 (1.00)

The following conditions exist:

-- RPV level cannot be maintained above TAF.
-- EOP-4 sheet 4 (Primary Containment Flooding) is executed to flood containment.
-- The containment level band specified by EOP-4 is between 62 ft and the MCWLL (84 ft).

Which of the following is the corresponding core level band to that specified by EOP-4?

a. -143 to +51 inches

b. -162 to +102 inches

c. -205 to +59 inches

d. -213 to +51 inches

QUESTION: 080 (1.00)

The scram discharge volume vent and drain valves did not close when a scram occurred.

Which of the following would be the adverse consequence?

a. There will be a primary leak into the containment.

b. Insufficient backpressure will cause excessive rod insertion speed.

c. The timer allowing scram reset after 10 seconds will not initiate.

d. The reactor pressure will be necessary to complete rod insertion.

QUESTION: 081 (1.00)

Isolation of a primary system leak is required by EOP-3, Secondary Containment and Radioactive Release Control, in order to limit radioactive discharge.

By definition, the term "Primary System" refers to any system:

a. for which the ASME "N" stamp is issued.

b. directly connected to the RPV that contains reactor coolant.

c. directly connected to the RPV that contains radioactive water.

d. connected directly to the RPV that has a reduced leak rate if RPV pressure is lowered.

QUESTION: 082 (1.00)

What is the PRIMARY reason that the Automatic Depressurization System is inhibited during an ATWS situation?

a. a considerable amount of energy could be put into the suppression pool well before it is necessary or required.

b. it would drive plant conditions above the RPV Saturation Temperature making RPV water level indication unreadable.

c. it would cause a large loss of RPV inventory and impose a severe thermal transient on the fuel.

d. once below the shutoff head of the low pressure ECCS systems, the injection water might cause a large power excursion.

QUESTION: 083 (1.00)

The following conditions exist:

- An ATWS is in progress
- Standby Liquid Control Pump "A" is running and injecting to RPV

Select the PRIMARY reason that EOP-1A, "RPV Control- ATWS," specifically prohibits starting both Standby Liquid Control Pumps to increase Boron injection rate.

a. The pumps are interlocked, "B" will not start if "A" is running.

b. The "B" system explosive valve will not fire if the "A" explosive valve has been fired.

c. Excess discharge pressure will lift both pump reliefs thus reducing Boron injection flow.

d. Excessive injection rate could result in power oscillations within the core.

QUESTION: 084 (1.00)

Which of the following conditions still constitutes "Adequate Core Cooling"?

NOTE: Only the injection sources stated are injecting. Regard each situation separately.

a. ATWS in progress, the feed system is maintaining level between -205 inches and -162 inches, MSIVs are open.

b. All rods in, MSIV/ADS valves are closed, RPV level is -200 inches and decreasing, RPV pressure is 200 psig.

c. All rods in, RCIC is injecting, 1 ADS valve is open, RPV level at -210 inches and increasing, MSIVs are closed.

d. ATWS in progress, CRD, RCIC and SLC (with Boron) are injecting, RPV level is -200 inches and increasing, MSIVs are open.

QUESTION: 085 (1.00)

Which of the following describes why Emergency Depressurization is directed, even without any injection source, in accordance with EOP-4, Alternate Level Control, when Reactor Water Level reaches -205 inches?

a. This is the minimum level that will generate sufficient steam to cool the uncovered core without induced steam flow.

b. Steam Cooling is not effective in removing decay heat when RPV level is above -205 inches.

c. Below this level there will be insufficient steam pressure to automatically open the ADS valves.

d. This level provides the maximum cooling effect per pound mass of steam available in the core.

QUESTION: 086 (1.00)

During power ascension the following plant conditions are noted to occur over a 3 minute period.

-- Reactor pressure	- decreased to 800 psig, now stable.
-- Reactor Water Level	- +25 inches trending to normal.
-- Reactor power	- decreased 5%, now stable at 50%
-- Generator output	- decreased to 550 Mwe from 600 Mwe.
-- No SCRAM	- No RPS actuations have occurred.

Which of the following is IMMEDIATELY required?

a. Increase power with recirculation flow.

b. Perform a reactor shutdown to hot standby.

c. Trip the main turbine.

d. Shut the MSIVs.

QUESTION: 087 (1.00)

Which of the following is the MINIMUM action required when a review of operating logs, one day after a transient, shows that a Safety Limit has been violated?

a. Obtain NRC approval to remain at power.

b. Restore safety limit to allowable value within 2 hours.

c. Place the reactor in Hot shutdown.

d. The reactor must be scrammed immediately.

QUESTION: 088 (1.00)

Intentional rod motion to SHORTEN a stable reactor period is NOT permitted if:

a. the existing period is 30 seconds or less.

b. power is on IRM range 8.

c. the existing period is 60 seconds or less.

d. the period is negative.

QUESTION: 089 (1.00)

The following conditions exist:

-- A Loss of all Site AC power has occurred..
-- No emergency diesels started.
-- All control rods are fully inserted.
-- Reactor level is -150 inches and decreasing.
-- RCIC is injecting.

The earliest that ADS/SRVs may be opened for emergency depressurization is:

a. at -162 inches.

b. at -193 inches.

c. at -205 inches

d. Only when RCS pressure drops below 600 psi.

QUESTION: 090 (1.00)

Identify the PRIMARY reason that reactor power goes down when reactor water level is deliberately lowered during a failure to scram (ATWS) event.

a. Further concentration of boron will result thus lowering the reactor power level.

b. Decreased reactor pressure will add negative reactivity due to reduced moderator density.

c. Increased core voiding will result from a decrease in natural circulation driving head and core flow.

d. Increased reactor water temperature will result, adding negative reactivity due to reduced moderator density.

QUESTION: 091 (1.00)

The following conditions exist:

-- The reactor has been scrammed .
-- There is a coolant leak into containment.
-- Plant parameters are approaching limits that require emergency depressurization.
-- Emergency depressurization is anticipated and main turbine bypass valves have been opened in accordance with EOP-1.

Conditions degrade and require immediate Emergency RPV Depressurization per EOP-4. Select the required action.

a. Open 7 ADS/SRVs and leave the bypass valves open.

b. Open 7 ADS/SRVs and close the bypass valves.

c. Close the bypass valves and open 7 ADS/SRVs valves.

d. Continue depressurization using only bypass valves.

QUESTION: 092 (1.00)

The reactor is shutdown with the RPV head in place.

Which of the following require that reactor water level be maintained above +75 inches using shutdown level instruments?

a. Anytime that no shutdown cooling loops are in operation but both are operable.

b. One recirc pump running with one shutdown cooling loop at less than full flow.

c. One recirc pump running with no shutdown cooling loops in operation.

d. Anytime that no reactor recirculation pumps are running and no shutdown cooling loops are available.

QUESTION: 093 (1.00)

With the reactor initially at 100% power, the following conditions have occurred:

-- Condenser vacuum started decreasing.
-- Power has been reduced using recirculation flow only.
-- Vacuum is 26" Hg and still decreasing.

Under these conditions, why is operation of the condenser air removal pumps prohibited?

a. The main steam line radiation monitor trip is bypassed when the mode switch is in RUN.

b. The condenser air removal pump discharge bypasses the Off-Gas system.

c. They cannot remove sufficient gases when greater than 15% power.

d. The vacuum reliefs will open to limit vacuum to 25" Hg.

QUESTION: 094 (1.00)

A recirculation pump was inadvertently tripped and is to be restarted. The temperature and flow requirements for starting an idle recirculation loop have been verified.

Select the MAXIMUM time allowed to start the pump before the temperature and flow requirements must be verified again.

a. 5 min.

b. 10 min.

c. 15 min.

d. 30 min.

QUESTION: 095 (1.00)

A recirculation pump was inadvertently tripped and the plant is operating in single loop.

The reactor and recirculation loop differential temperature requirements for single loop operation must be verified:

a. prior to increasing operating loop flow if below 50% of rated loop flow.

b. prior to decreasing operating loop flow if above 50% of rated thermal power.

c. every 15 minutes when the flow control valve is not at minimum in the idle loop.

d. every 15 minutes when the idle loop is isolated.

QUESTION: 096 (1.00)

All DC control power is lost to a 4160 Volt Circuit Breaker of an ECCS pump. (Assume ITE/ABB breaker)

Which of the following manual operations may be physically performed at the breaker without use of any breaker tools?

The breaker may be:

a. closed only (if open).

b. closed (if open) then tripped and closed again.

c. tripped open only.

d. tripped open, closed and tripped open again.

QUESTION: 097 (1.00)

With the reactor initially at 100% power, all Reactor Plant Component Cooling water is lost.

Select the MINIMUM required operator action.

a. Scram only.

b. Scram and shift both recirculation pumps to slow speed.

c. Rapid shutdown and shift both recirculation pumps to slow speed.

d. Scram, trip and isolate both recirculation pumps.

QUESTION: 098 (1.00)

While at 100% power, if only the upper recirculation pump seal fails on one pump, the expected increase in leak rate to the drywell (equipment sump) will be approximately:

a. 0.5 gpm

b. 5.0 gpm

c. 50 gpm

deleted

d. 60 gpm

QUESTION: 099 (1.00)

The following conditions exist:

-- The reactor has scrammed due to high pressure.
-- One Safety Relief Valve (SRV) lifted.

Which of the following would indicate that the SRV did not close?

The SRV discharge line temperature stabilizes at:

a. 212 deg F.

b. 240 deg F.

c. 305 deg F.

d. 625 deg F.

QUESTION: 100 (1.00)

The following conditions exist:

-- The plant has experienced a station blackout (loss of Div 1, Div 2 and Div 3 ENS Buses).
-- The Div 3 Diesel Generator was started and is running normally.
-- Emergency use of Div 3 for decay heat removal and RPV level control is being implemented.

Which of the following describes the general flowpath for this cooling mechanism?

a. CST - HPCS pump - RHR "A" heat exchangers - RPV - shutdown cooling drains to Suppression pool.

b. Suppression pool - HPCS pump - RPV - shutdown cooling to loop "A" RHR heat exchangers - test return to suppression pool.

c. CST - HPCS pump - RPV - shutdown cooling to loop "B" RHR heat exchanger - test return to Suppression pool.

d. Suppression pool - HPCS pump - RHR "A" heat exchanger - RPV - shutdown cooling drains to suppression pool.

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

A N S W E R   K E Y

MULTIPLE CHOICE

001 ~~d~~ a

002 c

003 b

004 a

005 c

006 c or b

007 a

008 d

009 b

010 a

011 b

012 d

013 b

014 c

015 d

016 a

017 b

018 b

019 a

020 b

021 c

022 a

023 a

024 a

025 c

026 d

027 b or a

028 b

029 c

030 b

031 d

032 b

033 d

034 c

035 c

036 d

037 a

038 c

039 c

040 c

041 b

042 a

043 a

044 b

045 c

ANSWER KEY

046 d or C
047 b
048 b
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050 a
051 b
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061 c
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063 a
064 d
065 a
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068 b
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072 c
073 c
074 d
075 d
076 c
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080 a
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085 a
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A N S W E R   K E Y

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093 b

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097 d

~~098~~ a * DELETED AFTER EXAM REVIEW

099 c

100 b

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

RO

Nuclear Regulatory Commission
Operator Licensing
Examination

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U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 4

CANDIDATE'S NAME: ______________________

FACILITY: River Bend 1

REACTOR TYPE: BWR-GE6

DATE ADMINISTERED: 95/08/18

INSTRUCTIONS TO CANDIDATE:

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

TEST VALUE	CANDIDATE'S SCORE	%	
~~100.00~~ 98.00	______ FINAL GRADE	___ %	TOTALS

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

____
Candidate's Signature

A N S W E R S H E E T

Multiple Choice (Circle or X your choice)

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

MULTIPLE CHOICE

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A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

046	a	b	c	d	___	069	a	b	c	d	___
047	a	b	c	d	___	070	a	b	c	d	___
048	a	b	c	d	___	071	a	b	c	d	___
049	a	b	c	d	___	072	a	b	c	d	___
050	a	b	c	d	___	073	a	b	c	d	___
051	a	b	c	d	___	074	a	b	c	d	___
052	a	b	c	d	___	075	a	b	c	d	___
053	a	b	c	d	___	076	a	b	c	d	___
054	a	b	c	d	___	077	a	b	c	d	___
055	a	b	c	d	___	078	a	b	c	d	___
056	a	b	c	d	___	079	a	b	c	d	___
057	a	b	c	d	___	080	a	b	c	d	___
058	a	b	c	d	___	081	a	b	c	d	___
059	a	b	c	d	___	082	a	b	c	d	___
060	a	b	c	d	___	083	a	b	c	d	___
061	a	b	c	d	___	084	a	b	c	d	___
062	a	b	c	d	___	085	a	b	c	d	___
063	a	b	c	d	___	086	a	b	c	d	___
064	a	b	c	d	___	087	a	b	c	d	___
065	a	b	c	d	___	088	a	b	c	d	___
066	a	b	c	d	___	089	a	b	c	d	___
067	a	b	c	d	___	090	a	b	c	d	___
068	a	b	c	d	___	091	a	b	c	d	___

ANSWER SHEET

Multiple Choice (Circle or X your choice)

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

092 a b c d ____

093 a b c d ____

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

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.

2. After the examination has been completed, you must 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. This must be done after you complete the examination.

3. Restroom trips are to be limited and only one applicant at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.

4. Use black ink or dark pencil ONLY to facilitate legible reproductions.

5. Print your name in the blank provided in the upper right-hand corner of the examination cover sheet and each answer sheet.

6. Mark your answers on the answer sheet provided. USE ONLY THE PAPER PROVIDED AND DO NOT WRITE ON THE BACK SIDE OF THE PAGE.

7. Before you turn in your examination, consecutively number each answer sheet, including any additional pages inserted when writing your answers on the examination question page.

8. Use abbreviations only if they are commonly used in facility literature. Avoid using symbols such as < or > signs to avoid a simple transposition error resulting in an incorrect answer. Write it out.

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

10. Show all calculations, methods, or assumptions used to obtain an answer to any short answer questions.

11. Partial credit may be given except on multiple choice questions. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.

12. Proportional grading will be applied. Any additional wrong information that is provided may count against you. For example, if a question is worth one point and asks for four responses, each of which is worth 0.25 points, and you give five responses, each of your responses will be worth 0.20 points. If one of your five responses is incorrect, 0.20 will be deducted and your total credit for that question will be 0.80 instead of 1.00 even though you got the four correct answers.

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

14. When turning in your examination, assemble the completed examination with examination questions, examination aids and answer sheets. In addition, turn in all scrap paper.

15. Ensure all information you wish to have evaluated as part of your answer is on your answer sheet. Scrap paper will be disposed of immediately following the examination.

16. To pass the examination, you must achieve a grade of 80% or greater.

17. There is a time limit of four (4) hours for completion of the examination.

18. When you are done and have turned in your examination, leave the examination area (EXAMINER WILL DEFINE THE AREA). If you are found in this area while the examination is still in progress, your license may be denied or revoked.

QUESTION: 001 (1.00)

An ATWS has occurred and the following conditions exist:

-- Reactor power	- 20% on APRMs
-- Reactor water level	- 35 inches
-- Drywell pressure	- 1.1 psig
-- All scram valves	- open
-- SDV vent and drain valves	- closed
-- Mode switch	- in SHUTDOWN
-- SDV water level	- high level scram signal in

Which of the following describes resetting of the scram to allow draining of the Scram Discharge Volume under these conditions?

a. The scram can be reset by placing the mode switch in STARTUP and the discharge volume high water level bypass keylock switch in "BYPASS".

b. The scram can only be reset 10 seconds after the mode switch is placed in SHUTDOWN.

c. The scram can be reset by placing the Scram Discharge Volume Bypass switch in "BYPASS."

d. The scram cannot be reset until EOP-0005 enclosure 12 is directed by the CRS.

QUESTION: 002 (1.00)

A licensed operator has been assigned as the At-The-Controls Operator for the following periods:

-- Friday 1600 to 0400
-- Saturday 1200 to 2400
-- Sunday 0800 to 1600
-- Monday 0800 to 1600
-- Tuesday 0800 to 2400
-- Wednesday 0800 to 2000

(Exclude turnover time)

On which day did he first violate the Overtime Guidelines?

a. Sunday

b. Monday

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c. Tuesday

d. Wednesday

QUESTION: 003 (1.00)

With the SS, CRS, STA, ATC operator ineligible as members, the fire brigade shall consist of at least:

a. four members not including one NEO and one other person required for safe shutdown of the plant.

b. four members including one other person not required for safe shutdown of the plant.

c. five members not including one NEO nor other personnel required for safe shutdown of the plant.

d. five members including one NEO and other personnel required for safe shutdown of the plant.

QUESTION: 004 (1.00)

Which of the following satisfies the MINIMUM requirements for MAINTAINING a valid active Reactor Operator license?

Once each calendar quarter the license holder must perform as a

a. Unit Operator or At-The-Controls Operator for 40 hours.

b. Unit Operator or At-The-Controls Operator for five 8-hour shifts including a complete plant tour.

c. Unit Operator or At-The-Controls position for five 12-hour shifts.

d. Reactor Operator At-The-Controls position for six 8-hour shifts including a plant tour.

QUESTION: 005 (1.00)

When performing protective tagging, who has the PRINCIPLE responsibility to ensure that proper isolation has been performed AND proper tagging is installed for employee protection?

a. the individual workers.

b. the person hanging the tags.

c. the person in charge of the work.

d. the person approving the clearance.

QUESTION: 006 (1.00)

Which of the following is a requirement for using a Human Tag?

a. V. P. Nuclear approval must be obtained.

b. Duration of the evolution must be less than one hour.

c. The supervisor of the job is responsible for ensuring that no one operates the device.

d. No more than three isolation points may be held by Human Tags at one time for a given maintenance activity.

QUESTION: 007 (1.00)

A Health Physics Technician has labeled a check valve as a "Hot Spot".

The MINIMUM contact dose rate that an operator should expect from this check valve is greater than:

a. 50 mRem

b. 100 mRem

c. 200 mRem

d. 500 mRem

QUESTION: 008 (1.00)

You are performing an in-plant surveillance test and leave the work area during the surveillance for a break.

In order to restart the surveillance test, in addition to notifying he OSS/CRS, you are required to verify at a MINIMUM that:

a. Only the prerequisites and all precautions and limitations are met.

b. Only that previously performed steps are in the same state.

c. Only that the prerequisites are still met.

d. Only that the prerequisites are met and previously performed steps are in the same state.

QUESTION: 009 (1.00)

You can verify that an operator aid has been approved if it contains:

a. the Department Foreman's signature.

b. the signature of an OSS or CRS.

c. the equipment lable request number.

d. a work document number.

QUESTION: 010 (1.00)

Which of the following describes the MINIMUM information that must be recorded on new chart paper immediately following installation?

a. Chart recorder number.

b. Current time and date.

c. Chart recorder number and panel number.

d. Current time and date and chart recorder number.

QUESTION: 011 (1.00)

In accordance with River Bend Station administrative requirements, deviations to procedures which are taken to protect the health and safety of the public must be approved by:

a. the Manager-Operations.

b. any SRO with an active license.

c. the Shift Superintendent or Control Room Supervisor.

d. the Operations Superintendent with notification to the NRC Resident Inspector.

QUESTION: 012 (1.00)

Who would NORMALLY give permission to enter the At-The-Controls Area?

a. Shift Superintendent.

b. Operations Superintendent.

c. Vice-President Operations.

d. General Manager Plant Operations.

QUESTION: 013 (1.00)

Which of the following would be indicative of a hydrogen deflagration in the off-gas system?

a. Decreasing offgas post treatment readings.

b. Fluctuating preheater inlet pressure.

c. Decreasing temperatures in the charcoal adsorber beds.

d. Steadily decreasing flow rates out of the charcoal adsorber beds.

QUESTION: 014 (1.00)

With the plant in Mode 3, which of the following would prohibit drywell entry?

a. Two Drywell Unit Coolers tagged out.

b. Drywell hydrogen concentration at 3.8%.

c. Drywell ambient temperature of 110 degrees F.

d. Drywell oxygen concentration at 19.3%.

QUESTION: 015 (1.00)

In order to provide shielding, the TIP detectors are normally stored:

a. inside the reactor vessel and above the core.

b. inside the reactor vessel and below the core.

c. within the "in-shield" position inside the TIP room.

d. within the indexing mechanism.

QUESTION: 016 (1.00)

If all normal fill systems are unavailable, which of the following systems would first be used to fill the upper pool?

a. Standby Service Water

b. Normal Service Water

c. Circulating Water

d. Turbine Plant Component Cooling Water

QUESTION: 017 (1.00)

Which of the following describes a properly oriented fuel bundle?

a. The orientation boss on the fuel assembly handle points away from the control rod.

b. The channel spacer buttons are adjacent to the control rod and adjacent to each other.

c. Serial number on the handle is readable from the outside edge of the four bundle fuel cell.

d. The channel fasteners are located on the outside edge of the four bundle fuel cell.

QUESTION: 018 (1.00)

Given the following conditions:

-- The plant is operating at 75% power.
-- The Control Room Operator places the Outboard MSIV Positive Leakage Control System switch to OPERATE.
-- Special risk based surveillances are being systematically performed on all safety related instrumentation systems.

Which of the following describes why the Outboard MSIV Positive Leakage Control System will NOT initiate?

a. A LOCA signal on either high drywell pressure or low reactor water level is not present due to a surveillance.

b. The post LOCA 20 minute timer has not timed out due to a surveillance.

c. The required main steam line pressure and reactor pressure requirements have not been met.

d. All Main Steam Isolation Valves have not been fully closed.

QUESTION: 019 (1.00)

The 1CNM-P1A Condensate Pump was started at 1045 and immediately tripped. The pump was restarted at 1110 and tripped at 1130.

The soonest this condensate pump may be restarted is:

a. immediately.

b. at 1145.

c. at 1210.

d. only when the motor cools to ambient.

QUESTION: 020 (1.00)

Five minutes after a LOCA the following conditions exist:

-- Reactor water level is -90 inches and decreasing.
-- Drywell pressure is 2.2 psig and increasing.

Which of the following describes the status of Division I and Division II safety-related loads.

a. Only ECCS motor loads are sequenced onto the safety buses. Existing vital loads on the buses continue to operate.

b. All loads on the safety buses are shed and vital loads are then sequenced onto the buses.

c. Only ECCS motor loads are sequenced onto the safety buses. Existing vital loads are shed.

d. ECCS loads are not yet sequenced onto the safety buses. Existing loads on the buses continue to operate.

QUESTION: 021 (1.00)

Normal Station Transformer 1STX-XNS1C should not be used to supply:

a. 1E22*S004 (HPCS standby bus) because there is no sync check permissive in the fast transfer from normal to preferred source.

b. 1ENS*SWG1A (standby bus) because there is no sync check permissive in the fast transfer from normal to preferred source.

c. 1E22*S004 (HPCS standby bus) to preclude the possibility of heating the transformer beyond the cooling fan capacity.

d. 1ENS*SWG1B (standby bus) to preclude the possibility of heating the transformer beyond the cooling fan capacity.

QUESTION: 022 (1.00)

Unless the backup charger is connected to the HPCS bus, the Backup Charger Supply Breaker (1E22-CB10) from 1E22-PNLSOC1 should be locked in the open position.

This ensures that:

a. the backup battery charger is not overloaded.

b. the backup charger continuously meets Technical Specification operability requirements.

c. separation between redundant Safety-Related systems is maintained.

d. damage to the rectifier stack does not occur.

QUESTION: 023 (1.00)

Given the following conditions:

-- Reactor water level is -90 inches and decreasing.
-- Drywell pressure is 2.2 psig and increasing.
-- An outside fire has caused smoke in the Control Room.
-- The operator has attempted to manually place the Control Room ventilation in the purge mode.

Under these conditions the Control Room Smoke Removal Damper (AOD 107/108) will:

a. open and the Smoke Removal Fan will start.

b. open but the Smoke Removal Fan will be interlocked off.

c. remain closed and the Smoke Removal Fan will run on recirc.

d. remain closed and the Smoke Removal Fan will be interlocked off.

QUESTION: 024 (1.00)

If total feedwater flow drops below the RRS interlock level, the Reactor Recirculation pumps will downshift to slow speed.

What is the PRIMARY reason for this interlock?

a. pump cavitation.

b. flow control valve cavitation.

c. excessive axial thrust on the pump.

d. inaccurate wide range level indication.

QUESTION: 025 (1.00)

Which of the following signals will result in transferring BOTH Reactor Recirculation pumps to slow speed (LFMG) BUT may be manually bypassed?

Note: Consider only permanently installed bypass capabilities.

a. EOC RPT Logic trip.

b. RPV low level-Level 3.

c. Low feedwater flow for 15 seconds.

d. Loop "A" Recirc pump suction to steam line differential temperature is less than 8 deg F for 15 seconds.

QUESTION: 026 (1.00)

The following conditions exist:

-- The plant was operating at 100% power.
-- A loss of BOTH Reactor Protection System (RPS) buses has occurred

Which of the following describes the operators responsibilities regarding the Main Steam Isolation Valves (MSIV) for this failure?

The MSIV:

a. control switches must be placed in "Close" before the valves begin to drift closed.

b. low vacuum isolation must be bypassed before reenergizing the RPS buses.

c. control switches must be placed in "Close" to reset the MSIV isolation, once power is restored to RPS.

d. control switches must be placed in "Close" before reenergizing the RPS buses.

QUESTION: 027 (1.00)

Which of the following will cause a FULL MSIV and MSL drain isolation ONLY (BOP valves remain open) using the isolation pushbuttons on 1H13*P680,1B21H-S25A(B)(C)(D)?

a. "A" and "C" must be depressed at the same time OR "B" and "D" must be depressed at the same time.

b. "A" and "D" must be depressed at the same time OR "B" and "C" must be depressed at the same time.

c. "A" must be depressed and released, then "D" must be depressed.

d. "A" must be depressed and released, then "C" must be depressed.

QUESTION: 028 (1.00)

The HPCS Suppression Pool Suction Valve (F015) will automatically open:

a. when the HPCS CST Suction valve (F001) automatically goes closed with a HPCS initiation signal present.

b. anytime a low CST level or high suppression pool level exists.

c. only when a HPCS initiation signal is present AND either a low CST level OR high suppression pool level exists.

d. two seconds after the HPCS CST Suction valve (F001) automatically goes closed.

QUESTION: 029 (1.00)

The following conditions exist:

-- The plant has experienced a loss of all AC power (BLACKOUT)
-- The Reactor Core Isolation Cooling (RCIC) system is running and injecting to the reactor.

RCIC then receives a valid Div.1 and Div.2 isolation signal.

Which of the following valves will CLOSE under these conditions?

Note: Only consider direct actions from a system isolation signal.

a. Turbine steam supply warmup valve-- F076

b. Injection shutoff valve-- F013

c. Turbine steam supply inboard isolation valve -- F063

d. Pump suppression pool suction valve -- F031

QUESTION: 030 (1.00)

Given the following conditions:

-- The Reactor Water Cleanup (RWCU) system is operating in the normal mode.
-- The RWCU Keylock Bypass Switches (E31-S1A,B) on P632 and P642 have been placed in "Bypass".

Select the expected effect on the RWCU system.

a. The RWCU system isolation on initiation of the Standby Liquid Control System is defeated.

b. The RWCU system isolation from the Leak Detection System is defeated.

c. All RWCU isolations, except for high differential flow and high area temperature are defeated.

d. All RWCU system isolation signals are defeated.

QUESTION: 031 (1.00)

Which of the following plant systems are required to be operable for SECONDARY Containment Integrity?

a. Combustible Gas Control System.

b. MSIV Positive Leakage Control System.

c. Penetration Valve Leakage Control System.

d. Standby Gas Treatment System.

QUESTION: 032 (1.00)

Which of the following will start the Division I Standby Diesel Generator but will NOT result in closure of the generator output breaker.

(Assume normal ENS Bus voltage is exactly 4160 VAC.)

a. ENS bus voltage has been 3300 VAC for 5 seconds.

b. Reactor water level is steady at -135 inches.

c. The "Arm and Depress" pushbutton for Low Pressure Core Spray/RHR has been depressed.

d. Drywell pressure is steadily increasing and has reached 1.50 psig.

QUESTION: 033 (1.00)

The following conditions exist:

-- The Div 2 standby diesel generator is loaded and in parallel with bus 1ENS*SWG1B through the normal breaker.
-- A LOCA signal occurs.

Which of the following describes the effect of the standby diesel generator and bus 1ENS*SWG1B?

a. The normal breaker to the bus will open and the diesel generator will supply bus loads.

b. The normal breaker and diesel output breaker will open then after loads are stripped, the diesel breaker will reclose .

c. The diesel generator output breaker will open and cannot be closed until the LOCA is reset.

d. The diesel generator output breaker will remain closed in parallel operation with the bus.

QUESTION: 034 (1.00)

Which of the following will result in an automatic Control Room ventilation system isolation?

a. Two or more control room smoke alarms.

b. Any automatic start of the Standby Gas Treatment system.

c. Any control building area radiation monitor alarm.

d. Control building local intake high radiation signal.

QUESTION: 035 (1.00)

The following conditions exist:

-- The reactor is at 100% power.
-- The Off-Gas Post Treatment HI-HI-HI radiation alarm (P601/22A/A03) has occurred.
-- The Offgas System automatically isolated, (1N64-F060 Off-Gas Discharge to Vent valve is closed).

Which of the following actions is PROHIBITED?

a. Purge the Off-Gas system with service air.

b. Shift to the Standby Off-Gas Component.

c. Place steam seals on auxiliary steam.

d. Reduce power below 60%.

QUESTION: 036 (1.00)

A startup is in progress and the following conditions exist:

-- Reactor power	- 2% on APRMs
-- Reactor water level	- 35 inches
-- Reactor pressure	- 550 psig
-- Condenser vacuum	- 20 inches
-- Mode switch	- in STARTUP
-- MSIVs	- open

Which of the following describes the effect if RPS bus "A" is deenergized?

a. All MSIVs will close.

b. MSIVs "A" and "C" will close.

c. A Div 1 half scram will occur.

d. A full scram will occur.

QUESTION: 037 (1.00)

During power reduction, which of the following Reactor Protection System automatic scrams is bypassed by taking the mode switch from RUN to STARTUP ?

a. Turbine Control Valve Fast Closure

b. Turbine Stop Valve Closure

c. Reactor Water Level - High

d. Scram Discharge water level - High

QUESTION: 038 (1.00)

Which of the following describes the response of a control rod if the ball check valve in the drive mechanism is stuck closed?

The control rod will:

a. withdraw faster than normal.

b. insert faster than normal.

c. scram faster than normal.

d. scram slower than normal.

QUESTION: 039 (1.00)

Which of the following describes how operation of the Backup Scram Solenoid Valves result in a reactor scram?

When actuated, the Backup Scram Solenoid Valves:

a. relieve hydraulic control unit drive pressure to the exhaust header.

b. isolate the withdraw side of the control rod and vent the control rod insert side to the exhaust header.

c. relieve the withdraw side of the control rod to the exhaust header allowing reactor pressure to scram the rod.

d. isolate and vent the air supply to the hydraulic control units.

QUESTION: 040 (1.00)

Which of the following are DC powered AND must energize to operate during a scram?

a. Backup scram valves.

b. SDV vent pilot valves

c. SDV drain pilot valves.

d. Scram pilot solenoid valves.

QUESTION: 041 (1.00)

During a startup conducted within 24 hours of previous high power operation, unusually high control rod worths may be experienced.

The range of control rod positions where this condition is particularly likely to occur are notch positions:

a. 0 - 16

b. 8 - 20

c. 12 - 24

d. 16 - 32

QUESTION: 042 (1.00)

While at 100% power, which of the following describes the effect on control rods of losing power to Alternate Rod Insertion (ARI) channel A AND channel B?

ARI will:

a. not initiate either automatically or manually from the control room.

b. not initiate automatically but will initiate manually from the control room.

c. isolate the air supply but not vent the scram air header.

d. automatically initiate.

QUESTION: 043 (1.00)

During a normal control rod insertion, what prevents drive water from recirculating back into the cooling water header?

a. The cooling water supply header to the hydraulic control unit includes a check valve.

b. The system cooling water header pressure is greater than drive water pressure.

c. The cooling water header isolation valve to the hydraulic control unit (104) closes when "Insert" is selected.

d. The control rod drive stabilizing valve closes on an "Insert" signal thus isolating the cooling water header.

QUESTION: 044 (1.00)

Which of the following describes a requirement for conducting control rod coupling checks?

a. For all control rods, prior to reactor criticality after completing core alterations.

b. Anytime the control rod is withdrawn to the full out position.

c. Only the first time that each control rod reaches position 48 after a reactor startup.

d. The first time that each control rod reaches position 48 after a reactor startup and weekly thereafter.

QUESTION: 045 (1.00)

The APRM Channel A meter function switch is placed in the "COUNT" position.

Which of the following is the MINIMUM indication for the APRM to be considered operable independent of level requirements?

a. 45%

b. 50%

c. 55%

d. 60%

QUESTION: 046 (1.00)

Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow if core flow is less than 70% of rated core flow.

b. 5% of rated recirculation flow if core flow is greater than 70% of rated core flow.

c. 10% of rated recirculation flow if core flow is greater than 70% of rated core flow.

d. 15% of rated recirculation flow if core flow is less than 60% of rated core flow.

QUESTION: 047 (1.00)

Which of the following will prevent RCIC discharge to the CST through the test line return valves (F022, F059)?

a. The CST suction valve is open (F010).

b. RCIC minimum flow valve open (F019).

c. CST level is less than 6.5 inches.

d. Reactor vessel level is 55 inches.

QUESTION: 048 (1.00)

Which of the following will automatically trip the Condenser Mechanical Vacuum Pump?

a. Offgas Post Treatment HI-HI-HI radiation.

b. Loss of the Seal Water pump ARC-P2A(B).

c. Main steam line HIGH radiation.

d. Mode switch in RUN.

QUESTION: 049 (1.00)

The division 1 diesel generator is started using the local "EMERGENCY START" PUSHBUTTON:

Which of the following diesel generator shutdown signals is bypassed in this condition?

a. Generator differential current.

b. Low diesel lube oil pressure.

c. Local manual stop pushbuttons.

d. Diesel overspeed.

QUESTION: 050 (1.00)

Which of the following require loss of power to both divisions of their control logic in order to cause an automatic isolation?

a. Residual Heat Removal valves

b. Main Steam Isolation valves

c. Reactor Water Cleanup valves

d. Reactor Core Isolation Cooling valves

QUESTION: 051 (1.00)

An SRV control switch is placed on OFF.

How does this affect the pressure relief, safety, and ADS functions of the valve?

a. Only the pressure relief function will not operate.

b. Only the safety function will not operate.

c. Only the ADS function will not operate.

d. Both the pressure relief and the ADS functions will not operate.

QUESTION: 052 (1.00)

While in a refueling outage, which of the following requires Primary Containment Integrity to be established?

a. Both trains of Standby Gas Treatment become inoperable.

b. One LPRM detector will be replaced with a new one.

c. All source range detectors are discovered to be inoperable.

d. Irradiated fuel is to be moved in the lower fuel pool.

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QUESTION: 053 (1.00)

Which of the following will bypass ALL rod blocks caused by SRM "A"?

a. All IRMs on range 8.

b. Reactor mode switch in REFUEL.

c. SRM "A" detector fully withdrawn.

d. SRM "A" function switch NOT in operate.

QUESTION: 054 (1.00)

The following conditions exist:

-- The mode switch is in RUN.
-- IRM "A" becomes INOP due to HV low.
-- IRM "A" is NOT in BYPASS.

Which of the following will subsequently cause a half scram?

a. An IRM "A" downscale signal.

b. Placing IRM "A" in BYPASS.

c. Placing IRM "A" in BYPASS with APRM channel "A" in BYPASS.

d. Placing the mode switch in STARTUP.

QUESTION : 055 (1.00)

The following conditions exist:

- - A valid Automatic Depressurization System (ADS) initiation signal has been received.
- - ADS Logic Channel "A" (Div 1) FAILS TO actuate due to loss of power to ENB*PNL02A
- - ADS Logic Channel "B" (Div 2) actuates (energizes).

Select the expected response from the ADS SRVs.

a. No ADS SRVs open.

b. Only three ADS SRVs open.

c. Only four ADS SRVs open.

d. All ADS SRVs open.

QUESTION: 056 (1.00)

The following conditions exist:

- - A leak inside the drywell has occurred.
- - All RHR pumps are running.
- - The Automatic Depressurization System automatically actuated at
- - 105 seconds.
- - All ADS SRVs are open.
- - Reactor water level is now steady at -150 inches.
- - Drywell pressure is now 1.5 psig and decreasing.
- - Reactor pressure is 200 psig.

If both Div 1 and Div 2 ADS TIMER/LEVEL 3 SEAL-IN RESET buttons are depressed and then released, which of the following describes the result on the Automatic pressurization System?

The ADS SRVs will:

a. close and then reopen after 5 minutes plus 105 seconds.

b. close and then reopen after 105 seconds.

c. close and remain closed.

d. remain open.

QUESTION: 057 (1.00)

When in Operational Condition 4 with decreasing coolant temperatures, select the MAXIMUM reactor head flange temperature that REQUIRES verification every 30 minutes?

a. 60 deg F

b. 80 deg F

c. 200 deg F

d. 300 deg F

QUESTION: 058 (1.00)

The following conditions exist:

-- A plant startup is in progress per GOP-0001.
-- No control rods have yet been withdrawn.

Which of the following describes when the reactor system temperature and pressure must be determined to be to the right of the criticality limit line?

a. 15 minutes prior to withdrawal of control rods and once per 5 minutes during heatup.

b. 15 minutes prior to withdrawal of control rods and once per 30 minutes during heatup.

c. 30 minutes prior to withdrawal of control rods and once per 5 minutes during heatup.

d. 30 minutes prior to withdrawal of control rods and once per 30 minutes during heatup.

QUESTION: 059 (1.00)

Which of the following describes the PRIMARY reason why operators are cautioned not to open RWCU Drain to Main Condenser (1G33-F046) and RWCU Drain to Rad Waste (1G33-F035), at the same time?

a. May provide an unmonitored release pathway through radwaste.

b. The heat exchangers cannot remove sufficient heat to prevent system isolation.

c. The cleanup pumps may trip on low suction pressure.

d. Condenser vacuum may be reduced.

QUESTION: 060 (1.00)

With the Refueling Platform over the core, w..ich of the following, BY ITSELF, will initiate an RCIS rod bloc..?

a. Mode switch in STAFTUP.

b. Grapple not full down.

c. Service Platform Hoist loaded.

d. Any rod not fully inserted

QUESTION: 061 (1.00)

With irradiated fuel stored in the spent fuel pool (SFP), mechanical maintenance is preparing to move a 1400 lb. motor across the pool using the fuel building overhead crane:

Why will this move not be permitted?

a. Mechanical maintenance is NOT authorized to use the fuel building overhead crane with fuel in the SFP.

b. A 1400 lb. weight exceeds the load limit of the fuel building overhead crane.

c. No load in excess of 1200 lb. is permitted to travel over the fuel storage racks by procedure.

d. Reactor engineering support is required for any evolution involving irradiated fuel.

QUESTION: 062 (1.00)

The following conditions exist:

-- A LOCA condition has occurred based on Level 1 and Hi Drywell Pressure.
-- Water level has been recovered.
-- The "A" train of RHR was placed in suppression pool cooling with the Injection Valve, 1E12*F042A, overridden closed.

Which of the following will remove the Injection Valve, 1E12*F042A, override condition?

Note - consider each choice independently.

a. Clearing all LOCA conditions.

b. Raising reactor pressure above 487 psig.

c. Placing the injection valve control switch to CLOSE then OPEN.

d. Depressing the RHR Division I/LPCS Initiation Reset pushbutton.

QUESTION: 063 (1.00)

The following conditions exist:

-- RHR shutdown cooling loop A is in operation.
-- Reactor water level is 75 inches.

Under these conditions, which of the fol` ving has NO interlock to prevent draining the reactor vessel?

a. Suppression Pool Suction Valve 1E12*F004A.

b. Shutdown Cooling Suction Valve 1E12*F006A.

c. Test Return To Suppression Pool Valve 1E12*F024A.

d. Shutdown Cooling Outboard Isolation Valve 1E12*F008A.

QUESTION: 064 (1.00)

In order to prevent unacceptable vibration levels, while E12*MOV F048A, RHR HX Bypass valve, is less than 5% open, RHR Pump A flow should be kept above a MINIMUM flow of:

a. 2000 gpm.

b. 2500 gpm.

c. 3500 gpm.

d. 4500 gpm.

QUESTION: 065 (1.00)

The following conditions exist:

-- The reactor successfully scrammed.
-- RPV level dropped to -10 inches and is now stable at 30 inches.
-- EOP 1 RPV Control was entered briefly and has been exited.

During performance of the subsequent actions of AOP-0001, Reactor Scram, RPV level control becomes erratic and level again reaches -10 inches.

Select the REQUIRED action.

a. Re-enter EOP-1 at the beginning.

b. Re-enter EOP-1 at step stating "Restore and maintain RPV water level between 9.7 and 51 inches."

c. Re-enter EOP-1 only if RPV level is not quickly restored above 9.7 inches.

d. Restore and maintain RPV water level between 9.7 and 51 inches, EOP entry is not necessary.

QUESTION: 066 (1.00)

During execution of EOP-1 (RPV control), entry into EOP-1A ATWS is NOT necessary if:

a. the reactor is shutdown now, but could become critical later.

b. all control rods are inserted to position 02.

c. all control rods are inserted to position 00 except one.

d. all control rods are inserted to position 02 except one.

QUESTION: 067 (1.00)

The following conditions exist:

-- An ATWS is in progress.
-- Reactor power is 22%.
-- Reactor water level is -10 inches.
-- Reactor pressure is 960 psig.

Which of the following will be the first to be severely challenged and is of primary importance should a full MSIV closure occur?

a. Primary containment integrity.

b. Secondary containment integrity

c. Fuel integrity.

d. RPV integrity.

QUESTION: 068 (1.00)

When EOP-4, sheet 2, Emergency RPV Depressurization, permits defeating isolation interlocks in order to rapidly depressurize without SRVs, which of the following MSIV isolation signals may be bypassed?

a. Only the RPV low level 1 signal.

b. Only the RPV Low Level 1 and low main steam line pressure signal.

c. All MSIV isolation signals except main steam line high radiation.

d. All MSIV isolation signals.

QUESTION: 069 (1.00)

The following conditions exist:

- -- An SRV spuriously opened and stuck open.
- -- Reactor power has been reduced to 85%.
- -- EOP-2, Primary Containment Control, has been entered
- -- Suppression pool cooling is in service.
- -- Suppression pool temperature is increasing.
- -- Efforts to close the SRV have been unsuccessful.

Which of the following is the INITIAL required action as suppression pool temperature is increasing beyond 100 deg F?

a. Scram and enter AOP-0001 if temperature reaches 105 deg F.

b. Enter EOP-1, RPV Control, and scram if temperature reaches 110 deg F.

c. Emergency depressurize if temp exceeds the Heat Capacity Temperature Limit.

d. Reduce reactor pressure as necessary to maintain pressure below the Heat Capacity Temperature Limit.

QUESTION: 070 (1.00)

While conducting refueling operations, the following has occurred.

- You are the fuel handler.
-- A fuel bundle is dropped during withdrawal from the core.
-- The bundle is resting about 30 degrees off vertical.
-- No radiation alarms have been received, but bubbles are observed rising from the fuel bundle.

Select the REQUIRED immediate action.

a. Stop all refueling operations and immediately evacuate all persons from the containment and drywell.

b. Retrieve the bundle and place it in the nearest safe position then evacuate the fuel building, containment, and drywell.

c. Move the bundle to a safe location and await instructions from the Recovery Manager.

d. Start the Annulus Mixing and Standby Gas Treatment systems and stop further fuel movements if any radiation alarm is received.

QUESTION: 071 (1.00)

Given a Mollier diagram and assuming rising RPV temperatures and pressures:

Select the first set of plant conditions for which ALL reactor water level indicators should NOT be used.

a.	RPV pressure Drywell/Containment temperature	60 psig 200 degrees F
b.	RPV pressure Drywell/Containment temperature	90 psig 300 degrees F
c.	RPV pressure Drywell/Containment temperature	200 psig 400 degrees F
d.	RPV pressure Drywell/Containment temperature	1000 psig 180 degrees F

QUESTION: 072 (1.00)

The following conditions exist:

-- The reactor is in cold shutdown and vented.
-- One reactor recirculation pump is operating.
-- One shutdown cooling loop is in operation.
-- RPV level is 35 inches.

Subsequently, all shutdown cooling and forced circulation is lost and RPV level is being raised greater than 75 inches.

Select the MAXIMUM vessel level allowed without flooding the steam lines?

a. 75 inches

b. 105 inches

c. 140 inches

d. 196 inches

QUESTION: 073 (1.00)

Without operator action, if the Turbine Stop Valve closure scram fails to actuate, reactor protection is provided by the:

a. main steam line isolation scram.

b. recirculation pump trip (RPT) circuit.

c. reactor vessel low level (9.7 inches).

d. nuclear system high pressure scram.

QUESTION: 074 (1.00)

The following conditions exist:

- -- A loss of all AC power occurred at 6:45 am.
- -- Reactor pressure was initially 1030 psig.
- -- The RCIC system was manually started.
- -- There is a small coolant leak into the containment
- -- A cooldown was initiated at 7:00 am.

The following RPV pressures have been recorded.

TIME	RPV Pressure
7:00 am	798 psig
7:15 am	610 psig

Assuming a constant rate of temperature reduction, what is the latest time that the cooldown must be secured in order to prevent exceeding the Technical Specification for cooldown?

Note: Answer rounded to nearest minute.

a. 07:20 am

b. 07:35 am

c. 07:50 am

d. 08:05 am

QUESTION: 075 (1.00)

With the reactor initially at 100% power, a loss of instrument air to which of the following will NOT eventually result in an automatic REACTOR PROTECTION SYSTEM SCRAM?

a. Condensate and heater drain pump recirc valves.

b. Feedwater regulating valves.

c. Turbine steam seals and SJAE.

d. Scram inlet and outlet valves.

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QUESTION: 076 (1.00)

A Safety Relief Valve tailpiece vacuum breaker has failed in the open position during SRV operation.

Which of the following will result?

a. Direct pressurization of the containment air space each time the SRV is opened.

b. Steam bypassing the quenchers with a direct discharge path into the suppression pool.

c. An increase in drywell to containment differential pressure.

d. Suppression pool water being drawn up into the SRV discharge line after the SRV is closed.

QUESTION: 077 (1.00)

Which of following conditions will ALWAYS require entry into EOP-3 Secondary Containment and Radioactive release control?

a. HPCS equipment area temperature high.

b. Turbine building ventilation automatic shutdown.

c. Auxiliary Building floor drain sump overflowing.

d. SBGT system automatically started.

QUESTION: 078 (1.00)

Intentional entry into the Region A or B of the Power/Flow Map during normal operation is:

a. permitted if Reactor Core instability is not indicated.

b. permitted if followed by an immediate manual scram.

c. permitted if the requirements of REP-0023 are followed.

d. never permitted.

QUESTION: 079 (1.00)

With both reactor recirculation pumps initially operating in fast speed and the rod line at 101%, the following occurs:

-- The "A" recirculation pump unexpectedly trips.
-- No unstable neutron instrumentation is observed.
-- The unit is determined to be in Region B of the Power/Flow Map.

Which of the following is required ?

a. Close the "A" recirculation pump flow control valve.

b. Insert control rods in the reverse order of the control rod insertion sequence of AOP-0001 until out of Region B only .

c. Insert control rods in the reverse order of the rod sequence package until out of Region B and Region C.

d. Manually scram the reactor.

QUESTION: 080 (1.00)

The following conditions exist:

-- The reactor is critical with the mode switch in STARTUP.
-- The running Control Rod Drive (CRD) pump has tripped and cannot be immediately restarted.
-- The other CRD pump is out for maintenance.
-- The accumulator trouble alarm for the rod being moved has just been received.

A reactor scram is required:

a. immediately.

b. if any other accumulator trouble alarm occurs.

c. only if an adjacent accumulator trouble alarm occurs

d. if any valid control rod temperature alarm is received.

QUESTION: 081 (1.00)

The following conditions exist:

-- RPV level cannot be maintained above TAF.
-- EOP-4 sheet 4 (Primary Containment Flooding) is executed to flood containment.
-- The containment level band specified by EOP-4 is between 62 ft and the MCWLL (84 ft).

Which of the following is the corresponding core level band to that specified by EOP-4?

a. -143 to +51 inches

b. -162 to +102 inches

c. -205 to +59 inches

d. -213 to +51 inches

QUESTION: 082 (1.00)

The following conditions exist:

-- A reactor startup is in progress.
-- The mode switch is in STARTUP.
-- The main turbine is tripped.
-- A valid MSIV isolation has occurred.
-- The reactor did not scram (No ATWS).

Which of the following was the only signal that could have generated the MSIV isolation?

a. Low reactor water level

b. High main steam line radiation

c. High steam line tunnel temperature

d. Low main steam line pressure

QUESTION: 083 (1.00)

The scram discharge volume vent and drain valves did not close when a scram occurred.

Which of the following would be the adverse consequence?

a. There will be a primary leak into the containment.

b. Insufficient backpressure will cause excessive rod insertion speed.

c. The timer allowing scram reset after 10 seconds will not initiate.

d. The reactor pressure will be necessary to complete rod insertion.

QUESTION: 084 (1.00)

Isolation of a primary system leak is required by EOP-3, Secondary Containment and Radioactive Release Control, in order to limit radioactive discharge.

By definition, the term "Primary System" refers to any system:

a. for which the ASME "N" stamp is issued.

b. directly connected to the RPV that contains reactor coolant.

c. directly connected to the RPV that contains radioactive water.

d. connected directly to the RPV that has a reduced leak rate if RPV pressure is lowered.

QUESTION: 085 (1.00)

What is the PRIMARY reason that the Automatic Depressurization System is inhibited during an ATWS situation?

a. a considerable amount of energy could be put into the suppression pool well before it is necessary or required.

b. it would drive plant conditions above the RPV Saturation Temperature making RPV water level indication unreadable.

c. it would cause a large loss of RPV inventory and impose a severe thermal transient on the fuel.

d. once below the shutoff head of the low pressure ECCS systems, the injection water might cause a large power excursion.

QUESTION: 086 (1.00)

The following conditions exist:

- An ATWS is in progress
- Standby Liquid Control Pump "A" is running and injecting to RPV

Select the PRIMARY reason that EOP-1A, "RPV Control- ATWS," specifically prohibits starting both Standby Liquid Control Pumps to increase Boron injection rate.

a. The pumps are interlocked, "B" will not start if "A" is running.

b. The "B" system explosive valve will not fire if the "A" explosive valve has been fired.

c. Excess discharge pressure will lift both pump reliefs thus reducing Boron injection flow.

d. Excessive injection rate could result in power oscillations within the core.

QUESTION: 087 (1.00)

Which of the following conditions still constitutes "Adequate Core Cooling"?

NOTE: Only the injection sources stated are injecting. Regard each situation separately.

a. ATWS in progress, the feed system is maintaining level between -205 inches and -162 inches, MSIVs are open.

b. All rods in, MSIV/ADS valves are closed, RPV level is -200 inches and decreasing, RPV pressure is 200 psig.

c. All rods in, RCIC is injecting, 1 ADS valve is open, RPV level at -210 inches and increasing, MSIVs are closed.

d. ATWS in progress, CRD, RCIC and SLC (with Boron) are injecting, RPV level is -200 inches and increasing, MSIVs are open.

QUESTION: 088 (1.00)

During power ascension the following plant conditions are noted to occur over a 3 minute period.

-- Reactor pressure	- decreased to 800 psig, now stable.
-- Reactor Water Level	- +25 inches trending to normal.
-- Reactor power	- decreased 5%, now stable at 50%
-- Generator output	- decreased to 550 Mwe from 600 Mwe.
-- No SCRAM	- No RPS actuations have occurred.

Which of the following is IMMEDIATELY required?

a. Increase power with recirculation flow.

b. Perform a reactor shutdown to hot standby.

c. Trip the main turbine.

d. Shut the MSIVs.

QUESTION: 089 (1.00)

Intentional rod motion to SHORTEN a stable reactor period is NOT permitted if:

a. the existing period is 30 seconds or less.

b. power is on IRM range 8.

c. the existing period is 60 seconds or less.

d. the period is negative.

QUESTION: 090 (1.00)

The following conditions exist:

-- The reactor has been scrammed .
-- There is a coolant leak into containment.
-- Plant parameters are approaching limits that require emergency depressurization.
-- Emergency depressurization is anticipated and main turbine bypass valves have been opened in accordance with EOP-1.

Conditions degrade and require immediate Emergency RPV Depressurization per EOP-4. Select the required action.

a. Open 7 ADS/SRVs and leave the bypass valves open.

b. Open 7 ADS/SRVs and close the bypass valves.

c. Close the bypass valves and open 7 ADS/SRVs valves.

d. Continue depressurization using only bypass valves.

QUESTION: 091 (1.00)

The reactor is shutdown with the RPV head in place.

Which of the following require that reactor water level be maintained above +75 inches using shutdown level instruments?

a. Anytime that no shutdown cooling loops are in operation but both are operable.

b. One recirc pump running with one shutdown cooling loop at less than full flow.

c. One recirc pump running with no shutdown cooling loops in operation.

d. Anytime that no reactor recirculation pumps are running and no shutdown cooling loops are available.

QUESTION: 092 (1.00)

With the reactor initially at 100% power, the following conditions have occurred:

-- Condenser vacuum started decreasing.
-- Power has been reduced using recirculation flow only.
-- Vacuum is 26" Hg and still decreasing.

Under these conditions, which of the following is PROHIBITED to restore vacuum?

a. Place the standby SJAE in service.

b. Start the vacuum priming system.

c. Start one condenser air removal pump.

d. Shift to the Standby Offgas Component.

QUESTION: 093 (1.00)

With the reactor initially at 100% power, the following conditions have occurred:

-- Condenser vacuum started decreasing.
-- Power has been reduced using recirculation flow only.
-- Vacuum is 26" Hg and still decreasing.

Under these conditions, why is operation of the condenser air removal pumps prohibited?

a. The main steam line radiation monitor trip is bypassed when the mode switch is in RUN.

b. The condenser air removal pump discharge bypasses the Off-Gas system.

c. They cannot remove sufficient gases when greater than 15% power.

d. The vacuum reliefs will open to limit vacuum to 25" Hg.

QUESTION: 094 (1.00)

A recirculation pump was inadvertently tripped and is to be restarted. The temperature and flow requirements for starting an idle recirculation loop have been verified.

Select the MAXIMUM time allowed to start the pump before the temperature and flow requirements must be verified again.

a. 5 min.

b. 10 min.

c. 15 min.

d. 30 min.

QUESTION: 095 (1.00)

A recirculation pump was inadvertently tripped and the plant is operating in single loop.

The reactor and recirculation loop differential temperature requirements for single loop operation must be verified:

a. prior to increasing operating loop flow if below 50% of rated loop flow.

b. prior to decreasing operating loop flow if above 50% of rated thermal power.

c. every 15 minutes when the flow control valve is not at minimum in the idle loop.

d. every 15 minutes when the idle loop is isolated.

QUESTION: 096 (1.00)

All DC control power is lost to a 4160 Volt Circuit Breaker of an ECCS pump. (Assume ITE/ABB breaker)

Which of the following manual operations may be physically performed at the breaker without use of any breaker tools?

The breaker may be:

a. closed only (if open).

b. closed (if open) then tripped and closed again.

c. tripped open only.

d. tripped open, closed and tripped open again.

QUESTION: 097 (1.00)

With the reactor initially at 100% power, all Reactor Plant Component Cooling water is lost.

Select the MINIMUM required operator action.

a. Scram only.

b. Scram and shift both recirculation pumps to slow speed.

c. Rapid shutdown and shift both recirculation pumps to slow speed.

d. Scram, trip and isolate both recirculation pumps.

QUESTION: 098 (1.00)

The reactor is at 100% power and the Shift Superintendent has decided to evacuate the control room due to nausea from an airborne contaminant.

Which of the following is the required action for the feed pumps?

a. Allow the running pumps to operate as is.

b. At +10 inches rising, stop two feedwater pumps.

c. If HPCS or RCIC is injecting, stop all feedwater pumps at +10 inches rising.

d. If HPCS and RCIC are not operating, verify feedwater pump trip at +51 inches.

QUESTION: 099 (1.00)

What is the MINIMUM condenser vacuum allowing continued plant operation at power?

a. 22.3" Hg vacuum

b. 23.5" Hg vacuum

c. 24" Hg vacuum

d. 25" Hg vacuum

QUESTION: 100 (1.00)

The following conditions exist:

-- The Div 1 Diesel Generator started due to a spurious low reactor water level signal.
-- It did not tie on to the ENS bus.
-- The LOCA initiation condition has cleared.
-- It is desired to restore ALL diesel generator trips to service and load the diesel prior to shutdown.

Select the MINIMUM action necessary to restore all diesel generator trips.

a. Depress the D/G emergency start reset on P877.

b. Reset the LOCA signal on P601.

c. Reset the LOCA signal on P601 AND depress the D/G emergency start reset on P877.

d. The diesel generator must be shutdown.

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

A N S W E R K E Y

MULTIPLE CHOICE

001	d
~~002~~	~~d~~ e DELETED AFTER EXAM REVIEW T— 9/14/91
003	c
004	c
005	c
006	c or b
007	b
008	d
009	b
010	b
011	c
012	a
013	b
014	d
015	b
016	a
017	b
018	c
019	a
020	a
021	a
022	c
023	d
024	b or a
025	c
026	c
027	c
028	b
029	d
030	b
031	d
032	c
033	c
034	d
035	a
036	c
037	c
038	d
039	d
040	a
041	a
042	a
043	a
044	b
045	c

ANSWER KEY

046	b	069	a
047	d or C	070	a
048	c	071	c
049	b	072	b
050	b	073	d
051	a	074	b
~~052~~	~~b~~ DELETED AFTER EXAM REVIEW JM 9/11/95	~~075~~	~~d~~ DELETED AFTER EXAM REVIEW JM 9/11/95
053	a	076	c
054	d	077	c
055	d	078	d
056	b	079	c
057	b	080	b
058	b	081	b
059	d	082	c
060	a	083	a
061	c	084	d
062	b	085	d
063	c	086	c
064	c	087	b
065	a	088	d
066	c	089	a
067	a	090	a
068	d	091	d

ANSWER KEY

092	c
093	b
094	c
095	a
096	d
097	d
098	a
099	d
100	b

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

ATTACHMENT 4

FACILITY LICENSEE POST EXAMINATION COMMENTS

ENTERGY

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JOHN R. McGAHA, JR.
Vice President
Operations

September 6, 1995

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

RBF1-95-214
RBG-41935
RBEXEC-95-137

Gentlemen:

In accordance with NUREG-1021, Revision 7, Supplement 1(ES-402, Attachment 3), enclosed are four questions submitted for your review and consideration. The enclosed questions (and supporting documentation) are a result of the NRC administered licensing examination conducted at the River Bend Station the week of August 18, 1995. The chief examiner during the examination was Mr. Howard Bundy.

If you have any questions regarding the attached, please contact Mr. L. Grant Lewis at (504) 381-4752.

Sincerely,

Mike Sellman
for JR McGaha

JRM/JJF
enclosure

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1) Question #24 RO/#27 SRO

If total feedwater flow drops below the RRS interlock level, the Reactor Recirculation pumps will downshift to slow speed.

What is the PRIMARY reason for this interlock?

a. pump cavitation.
b. flow control valve cavitation.
c. excessive axial thrust on the pump.
d. inaccurate wide range level indication.

Answer: b

Reference: LOTM-7-5, page 12 of 38, section V.A.1
Recirculation System Enabling Objective 3.3
K/A 202002K108, (3.1/3.2)

Comment:

Both (a) and (b) are correct. LOTM-7-5 page 12 of 38, section V.A.1 does give the key's answer but does not mention anything about being the PRIMARY reason for the interlock. There is some conflicting information that supports answer (a). LOTM 34-6 (Feedwater Level Control System), page 5 of 16, B.3 specifically states that the low total feedwater flow limit is based on preventing cavitation of the Reactor Recirculation pumps due to inadequate subcooling. HLO-005-05 (lesson plan), page 10 of 27, 4.1.4 does not give a PRIMARY component. Also HLO-060-5 (lesson plan), page 9 of 26 "NOTE" reinforces answer (a) to the students.

Recommendation:

Accept both answer (a) and (b).

- Reactor Water Cleanup (Chapter 14),
- Floor and Equipment Drains (Chapter 49), and
- Nuclear Boiler Instrumentation (Chapter 3).

V. SYSTEM OPERATION

A. Normal Operation

1. Cavitation Interlocks

   a. Cavitation is prevented by inhibiting high speed operation with total feed water flow less than 25%.

   - If pumps are operating at high speed and feed flow drops below 25%, pumps auto transfer to low speed after a 15-second time delay.
   - Starting or transferring to high speed is inhibited.
   - White light illuminates above the cavitation reset pushbutton on P680 panel, indicating a "seal-in" condition.
   - This interlock downshifts the pumps because feed flow of less than 25 percent will not provide adequate NPSH to prevent cavitation of the FCVs during fast speed recirc pump operation.
   - This interlock may be bypassed by taking SW-127A(B), Low Total Flow Interlock Bypass switch, located on panel H22-B33-P001A(B) to BYPASS.

   b. If reactor level falls to level 3 (9.7"), auto transfer to slow speed is initiated. This also allows for more accurate wide range level reading. The above interlocks will downshift both pumps to slow speed.

   c. If both loops receive a main steamline temperature/recirc pump suction differential temperature signal of <8°F, auto transfer to slow speed will be initiated after a 15-second time delay.

differential pressure transmitter (1C33-FT-N003A,B,C,D) is sent to the Main Control Room where it passes through an SRU to be converted from current to voltage. The following relationship exists between ΔP and flow: Flow = constant x $\sqrt{\Delta P}$. Since the resultant flow signal is not linear, the signal then passes through a square root extractor (Figure 1). The square root extractor linearizes the signal such that flow indications are linear and the control system responds in a linear fashion.

When approximately 40% rated steam flow is sensed in all 4 steam lines, a close signal is sent to B21-F033 (steam line inboard drain) and B21-F069 (steam line outboard drain).

Each of the four flow signals is then se^t to a summing circuit whose output is a total steam flow signal. In addition, the four steam line flows are displayed on meters on P680 (pounds/hr x 10^6).

3. Feedwater Flow (Figure 4)

Feedwater flow is measured by a venturi type flow element (1C33-FEN001A,B) located in each of the two feedwater lines to the reactor vessel. Each venturi differential pressure signal is measured by a transmitter (1C33-FT-N002A,B). Similar to steam flow, the feedwater flow signals pass through an SRU and a square root extractor (Figure 1). The output signal from the square root extractor is sent to a summer where the two feedwater flow signals are summed to give the total feedwater flow. The two feedwater flows are also displayed on meters on P680 (figure 10).

Low total feedwater flow ($\leq$25% rated for 15 seconds) provides a signal to transfer the Reactor Recirculation pumps from fast to slow speed during downpower. This limit is based on preventing cavitation of the Reactor Recirculation pumps due to inadequate subcooling. It also prevents upshift from slow to fast speed during power ascension if this permissive is not met.

C. **Component Description**

1. Feed Flow/Steam Flow Summer(Figure 1 - K602)

The total feed water flow and steam flow signals are input into the feed flow/steam flow summer.

4.0 Description of controls and interlocks

OBJ #4a. 4.1 Pump Cavitation Interlocks

CB **NOTE:** List the cavitation interlocks on the board as each is discussed.

4.1.1 Cavitation is prevented by inhibiting high speed operation with total feed water flow less than 25%.

- If pumps are operating at high speed and feed flow drops below 25%, pumps auto transfer to low speed after a 15-second time delay.
- Starting or transferring to high speed is inhibited.
- White light illuminates above the cavitation reset pushbutton on P680 panel, indicating a "seal-in" condition.
- This interlock downshifts the pumps because feed flow of less than 25 percent will not provide adequate NPSH in the downcomer for fast speed recirc pump operation.
- This interlock may be bypassed by taking SW-127A(B), located on panel H22-B33-P001A (B) to BYPASS.

4.1.2 If main steamline temperature/recirc pump suction differential temperature falls to 8°F, auto transfer to slow speed will be initiated after a 15-second time delay.

- This interlock is provided because the low pressure area at the recirc pump suction might fall below saturation pressure of the coolant under such a low differential temperature condition
- This interlock may be bypassed by taking SW-125A(B located on panel H22-B33-P001A(B) to BYPASS.

4.1.3 If reactor level falls to level 3 (9.7") auto transfer to slow speed is initiated.

4.1.4 All of the above interlocks are specifically designed to prevent or limit cavitation at the:

- Jet pumps
- Recirculation pumps
- Flow Control Valves

linearize the signal.

TP-06 3.0.4 Two meters on P680-3B indicate feedwater flow is the two lines with a scale of pounds/hr x 10^6.

3.0.5 The two feedwater flow signals are input to a total feed flow summer to provide a total feedwater flow signal. The output of this summer is sent to:

- Total feedwater flow indication (steam flow/feed flow recorder P680-3B).
- Feed flow/steam flow summer
- Bailey alarm cards K618A/B send a low total feedwater flow signal to the Recirculation System. This signal will transfer recirculation pumps to LFMG during a downpower. It also prevents upshift from slow to fast speed during power ascension (if this permissive is not met).

NOTE: Ask students what the purpose of the downshift is and what the setpoint is? **ANS**: Purpose is to prevent pump cavitation due to inadequate subcooling (≤25% rated flow for 15 seconds).

4.0 Feedwater Level Control System Components

OBJ#3.1

4.1 Feed Flow/Steam Flow Summer

TP-04 4.1.1 The total feedwater flow and steam flow signals are input into the feed flow/steam flow summer.

4.1.2 Its purpose is to provide a method for the Feedwater Level Control System to anticipate Rx level changes and take appropriate corrective action.

4.1.3 The principle is simple. If feed flow matches steam flow, level should be relatively constant. If a match does not exist, a change in level can be expected.

4.1.4 The feedwater flow input is negative

2) Question #47 RO/#46 SRO

Which of the following will prevent RCIC discharge to the CST through the test line return valves (F022, F029)?

a. The CST suction valve is open (F010)
b. RCIC minimum flow valve open (F019)
c. CST level is less than 6.5 inches.
d. Reactor vessel level is 55 inches.

Answer: d

Reference: LOTM-20-4, RCIC, page 3, II.A.2 and table #1
RCIC objective 4 and 5.j
K/A 217000A301, (3.5/3.5)

Comment:

Answer (c) could also be correct. Operators know that RCIC pump suction is normally lined up to the CST as long as there is at least about 3 and a half feet in the CST. This lineup automatically shifts the suction path from the CST to the suppression pool when there is less than 3'5" in the CST which correlates to a trip instrument "0" on the back panels. When the swap of the suction valves takes place, the test return valves are interlocked with the suppression pool suction valve to go shut and will prevent RCIC discharge to the CST. Since 6.5" is less than 3'5" this would be a correct answer. The confusion is that the response choice is not clear if this is actual tank level or a trip instrument level. To get actual CST level, the operator calls the auxiliary control room which gives actual CST tank level in feet. The operator can also call up a computer point which is also in units of feet (actual tank level). See LOTM-20-4, page 3, II.A.2. Also see table 2 (LOTM-20-4, page 15 of 24). Also see Tech. Spec. table 3.3.3-2 note ** page 3/4 3-39 for tank level correlation. Also see Tech. Spec. table 3.3.5-2, page 3/4 3-57 for trip setpoint.

Recommendation:

Accept both answer (c) and (d)

C. General Description

The RCIC system is started automatically upon receipt of a low reactor water level signal (level 2) or manually by the operator. Water from the CST or suppression pool is pumped into the core by a turbine-driven pump powered by reactor steam.

D. Basic System Flow Path

The RCIC pump suction is normally lined up to the Condensate Storage Tank (CST). This provides an adequate supply of high purity water for system operation. Steam to operate the turbine is supplied via piping from Main Steam Line (MSL) A upstream of the inboard Main Steam Isolation Valve (MSIV). The RCIC pump discharges to the reactor vessel upper head spray nozzle. A backup source of water for the RCIC pump is available from the suppression pool. Shifting to this source of water under normal conditions requires deliberate operation of valves by the operator.

II. SYSTEM DETAILS

A. Detailed Flow Path (Figure 1)

1. Steam Flow

Steam for operation of the RCIC turbine is provided from MSL A inside the drywell upstream of the inboard MSIV (B21-F022A). The steam piping size is 8" up to the point that taps off to the RHR system, and then reduces to 4" to supply the RCIC turbine.

The steam piping to the turbine is kept hot to allow rapid starting of the turbine, therefore allowing rated RCIC system flow to be attained in less than 30 seconds when needed. This is accomplished by a normal valve lineup and piping and steam trap arrangement which maintains the steam piping at near normal operating temperature.

The turbine is designed for immediate starting with no warmup prior to operation at rated speed. Turbine exhaust steam is directed to the suppression pool for condensation via 12" piping.

A gland seal air system is provided to prevent steam leakage from the turbine glands, governor and throttle valves. Air is provided by a compressor to counteract steam leakage from the above points.

2. Water Flow

The RCIC pump suction is normally lined up to the CST. Suction automatically shifts to the suppression pool when:

- a low level exists in the CST (0"), or
- a high level exists in the Suppression Pool (+6.5") with a RCIC isolation signal not present.

NOTE: Tech Specs require RCIC suction to shift at ≤6.5", or 20'-3.5" actual Suppression Pool level. Current setpoint is 19'-10.9", which satisfies this requirement.

Table 2 (continued)

1H13*P601/21A ANNUNCIATORS

DIV II RCIC ISOL STM SPLY PRESS LOW	60 psig (3 sec TD)	RCIC System Auto Isolation: • RCIC turbine trip. • RCIC and RHR steam supply inboard isol. valve (1E51*F063) closes. • RCIC steam line warmup isol. valve (1E51*F076) closes. • RCIC pump min flow to suppression pool valve (1E51*F019) closes. • RCIC injection isol. valve (F013) closes.
RCIC WARMUP LINE ISO VLV E51-F076 NOT FULLY CLOSED	Valve not closed	None.
RCIC TURBINE STEAM SPLY WATER DRAIN TRAP LVL HI	0" Increasing (mid-range on trap)	RCIC steam supply drain trap bypass valve (1E51*F054) opens.
RCIC TURB TRIP PMP SUCT PRESS LOW	20" Hg vac Decr. (0.5 second TD)	RCIC Turbine Trip.
RCIC SUCT XFER CST LEVEL LOW	0" on meter	• RCIC pump suppression pool suction valve (1E51*F031) opens. • RCIC pump CST suction valve (1E51*F010) closes. • RCIC test bypass valve to CST (1E51*F022) closes. • RCIC test return valve to CST (1E51*F059) closes.
RCIC ISOLATION RCIC RM HI AMB OR VENT DIFF TEMP	Ambient 182°F Vent Diff 96°F ΔT	RCIC System Auto Isolation: • RCIC turbine trips. • RCIC steam supply valves (1E51*F063 and 1E51*F064) close. • RCIC inject isol valve (1E51*F013) closes. • RCIC pump suppression pool suction valve (1E51*F031) closes. • RCIC min flow valve to suppression pool (1E51*F019) closes. • RCIC warmup line shutoff valve (1E51*F076) closes.

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (continued)		
2. Division III		
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	a. 4.16 kv Basis - 3045 ± 153 volts	3045 ± 214 volts
	b. 3 ± 0.3 sec. time delay	3 ± 0.33 sec. time delay
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3777 ± 30 volts	3777 ± 75 volts
	b. 60 ± 6 sec. time delay (w/o LOCA)	60 ± 6.6 sec. time delay
	c. 3 ± 0.3 sec. time delay (w/LOCA)	3 ± 0.33 sec. time delay

*See Bases Figure B 3/4 3-1.

**(Bottom of CST is at EL 95'1".) The levels are measured from the instrument zero level of EL 98'6".

#(Bottom of suppression pool is at EL 70'.) The levels are measured from the instrument zero level of EL 89'9".

##These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low Level 2	≥ -43 inches*	≥ -47 inches
2. Reactor Vessel Water Level - High Level 8	≤ 51 inches*	≤ 52 inches
3. Condensate Storage Tank Level - Low	≥ 0 inches	≥ -4.5 inches
4. Suppression Pool Water Level - High	≤ 6.5 inches	≤ 8 inches
5. Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.

3) Question #52 RO/#49 SRO

While in a refueling outage, which of the following requires Primary Containment Integrity to be established?

a. Both trains of Standby Gas Treatment becomes inoperable.
b. One LPRM detector will be replaced with a new one.
c. All source range detectors are discovered to be inoperable.
d. Irradiated fuel is to be moved in the fuel pool.

Answer: b

Reference: TS 3.6.1.2, page 6-2 amendment 35 and definition 1.7 Core Alterations
Primary containment objective 9A and 9B
K/A 223001G011, (3.3/4.2)

Comment:

No correct answer. Answer (b) assumes that replacing an LPRM is a core alteration. For our plant (BWR/6), the LPRM resides in a dry tube and if removed during a refuel outage, would have only a negligible (if any) effect on core reactivity. Replacing an LPRM is not a core alteration. See attached NRC safety evaluation of T.S.amendment No. 29, dated October 12,1988

Recommendation:

Throw out the question since there is no correct answer.

Attached to: RBC-37672

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated August 5, 1988, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would modify the Technical Specifications (TSs) to revise the definition of core alteration to exclude the normal movement (including replacement) of local power range monitors (LPRMs) from this definition.

2.0 EVALUATION

Technical Specification Definition 1.7, CORE ALTERATION, currently does not consider normal movement of the source range monitors, intermediate range monitors, traversing in-core probes, or special moveable detectors to be considered a core alteration. This change request would provide the same exclusion for LPRMs.

River Bend Station is a BWR/6 boiling water reactor which incorporates certain design changes compared to earlier boiling water reactors. One of these changes is the introduction of a dry tube that houses the LPRM strings. The dry tubes extend from the bottom of the reactor pressure vessel vertically to the top of the core. Thus, removal and installation of the LPRMs from underneath the reactor pressure vessel can be accomplished without the removal of the reactor vessel head and fuel does not need to be moved from around the dry tube for maintenance or replacement of LPRMs. The LPRM strings are only removed from the core when they are being replaced and they have no normal drive mechanisms. Based on the above discussion, the staff concludes that the exclusion of the LPRMs in the definition of core alteration is acceptable.

With the modification of the definition of core alteration discussed above, the footnote excepting replacement of LPRM strings applicable to Action 3 and Action 9 of Table 3.3.1-1 is no longer necessary. The staff concludes that deletion of the footnote is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 12, 1988

Principal Contributor: W. Paulson

4) Question #75/#71 SRO

With the reactor initially at 100% power, a loss of instrument air to which of the following will NOT eventually result in an automatic REACTOR PROTECTION SYSTEM SCRAM?

a. Condensate and heater drain pump recirc valves.
b. Feedwater regulating valves.
c. Turbine steam seals and SJAE.
d. Scram inlet and outlet valves.

Answer: d

Reference: AOP-0008, R7A, section 4.0
AOP-0008 objective 3.a
K/A 295019K201, (3.8/3.9)

Comment:

The words "will not eventually" leaves this question with no correct answer. Answer (a) does not send a direct RPS trip but will eventually cause a feedpump trip and eventually a RPV level 3 scram (see AOP-0008, Rev.7A page 3 of 16, section 3.2). Answer (b) will cause the feedwater regulating valves to lock up and fail "as is". Changes in power, pressure, temperature, and other non-constant external and internal forces (even fuel depletion) will cause a steam flow to feed flow mismatch and you'll get either a high or low RPV level scram. Also the air pressure that is locked up, keeping the feedwater regulating valves in a steady, constant position, will eventually bleed slowly. Answer (c) will result in a loss of condenser vacuum which does not send a direct RPS trip but will eventually cause a turbine trip and a resultant Rx scram (see AOP-0008, Rev. 7A page 3 of 16, section 3.5). Answer (d) will cause the rods to go in, and will cause either the scram discharge volume to fill faster than it can drain (assuming that the scram discharge volume vent & drain valves did not fail closed since the air line is common to the scram inlet and outlet valves which lost air) and cause a RPS scram or because the turbine is on the line and Rx power level is going down due to rods going in will cause Rx pressure to go below 849 psig (eventually because of other steam loads) and cause the MSIV's to go closed and cause a RPS scram (see LOTM-5 Fig. 12, Tech.Spec table 2.2.1-1(6)&(9), pages 2-4&5, and Tech.Spec.table 3.3.2-2(2.c), page 3/4 3-19).

Recommendation:

Throw out the question since there is no correct answer.

1.0 PURPOSE/DISCUSSION

1.1 The purpose of this procedure is to provide guidance to the operators in the event that Instrument Air System air pressure is lowering or lost.

1.2 A total loss of instrument air pressure may be caused by a break in the instrument air header, or by a loss of all air compressors. A multitude of actions occur as a result of low instrument air supply pressure. These actions occur at various times depending upon the rate of the instrument air pressure drop. The actions listed in 3.0, Automatic Actions, are listed in decreasing order of significance to the Nuclear Steam Supply System.

2.0 SYMPTOMS

2.1 Lowering Instrument Air Header Pressure.

2.2 Amber indicating lights for compressor 1IAS-C1A, C1B and/or C1C.

2.3 Various AOV's will fail (see Attachment 1) in a random manner.

3.0 AUTOMATIC ACTIONS

3.1 Control rods individually scram as the scram valves fail open. The Scram Discharge Volume vent and drain valves fail closed. CRD flow control valves fail closed.

3.2 Condensate and heater drain pumps recirc valves will open and "starve" the reactor feed pumps; causing them to trip

3.3 The Feedwater Reg Valves will lock up and fail "as is" on low air pressure (85 psig).

3.4 All normal HVAC will fail due to closure of AOD's.

3.5 Loss of steam seals and SJAE will result in a loss of condenser vacuum.

3.6 Drywell and containment drains will isolate.

4.0 IMMEDIATE OPERATOR ACTIONS

4.1 Make a plant wide Gaitronics announcement to cease non-essential use of air.

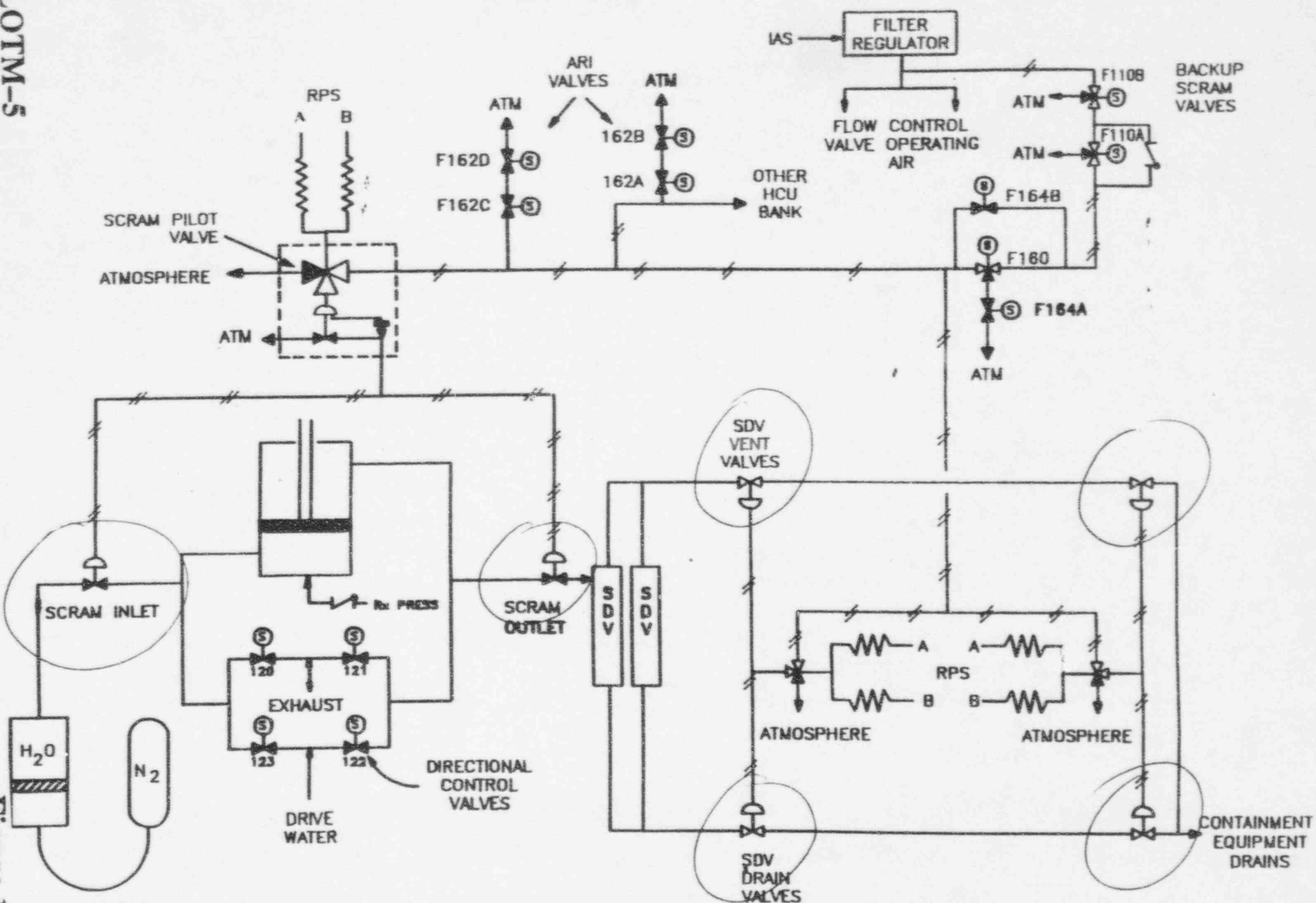
4.2 If any of the following occurs, Scram the Reactor and enter AOP-0001 REACTOR SCRAM:

4.2.1 When individual rod movement is observed.

4.2.2 When the instrument air header pressure decreases to 65 psig (1IAS-PI105) on 1H13*P870.

5.0 SUBSEQUENT OPERATOR ACTIONS

5.1 Monitor air header pressure and if it lowers to 50 psig, verify closed or close the MSIV's.



SCRAM AIR HEADER

Figure 12

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisior of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Two Recirculation Loop Operation		
a) Flow Biased	≤ 0.66 W+48%, with a maximum of	≤ 0.66 W+51%, with a maximum of
b) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	< 0.66 W+42.7%, with a maximum of	< 0.66 W+45.7%, wi a maximum of
b) High Flow Clamped	< 111.0% of RATED THERMAL POWER	< 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 9.7 inches above instrument zero*	≥ 8.7 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	≤ 51.0 inches above instrument zero*	≤ 52.1 inches abov instrument zero
6. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed

*See Bases Figure B 3/4 3-1.

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
8. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
9. Scram Discharge Volume Water Level - High		
a. Level Transmitter - LISN601A and B	$\leq$ 49"	$\leq$ 53"
LISN601C and D	$\leq$ 49"	$\leq$ 51.7"
b. Float Switches - LSN013A and B	$\leq$ 48.76"	$\leq$ 53.50"
LSN013C and D	$\leq$ 46.88"	$\leq$ 49.00"
10. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 530 psig	$\geq$ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-2
ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$-43 inches*	$\geq$-47 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Containment Purge Isolation Radiation - High	$\leq$ 1.3 R/hr	$\leq$ 1.57 R/hr
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$-143 inches*	$\geq$-147 inches
b. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
c. Main Steam Line Pressure - Low	$\geq$ 849 psig	$\geq$ 837 psig
d. Main Steam Line Flow - High		
1. Line A	$\leq$ 146 psid	$\leq$ 151 psid
2. Line B	$\leq$ 156 psid	$\leq$ 161 psid
3. Line C	$\leq$ 153 psid	$\leq$ 158 psid
4. Line D	$\leq$ 164 psid	$\leq$ 169 psid
e. Condenser Vacuum - Low	$\geq$ 8.5 inches Hg. vacuum	$\geq$ 7.6 inches Hg. vacuum
f. Main Steam Line Tunnel Temperature - High	$\leq$ 141°F	$\leq$148.5 °F
g. Main Steam Line Tunnel Δ Temperature - High	$\leq$ 57°F	$\leq$61 °F

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