UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report No.: 50-325/94-300

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket Nos.: 50-325 and 50-324

License Nos.: DPR-71 and DPR-62

Facility Name: Brunswick Steam Electric Plant

Examination Conducted: January 24-28, 1994

Chief Examiner: James H Moorman, III — 2-10-94
James H. Moorman, III — Date Signed

Accompanying Personnel: R. Baldwin, NRC Region II
T. Bettendorf, Battelle
K. Mikkelsen, Battelle

Approved By: Lawrence L Lawyer — 2/14/94
Lawrence L. Lawyer, Chief — Date Signed
Operator Licensing Section
Operations Branch
Division of Reactor Safety

SUMMARY

Scope:

NRC examiners conducted regular, announced operator licensing initial examinations and associated inspection activities during the periods January 10-14, 1994, and January 24-28, 1994. Examiners administered examinations under the guidelines of the Examiner Standards, NUREG-1021, Revision 7. Seven ROs and five SROs received written examinations and operating tests. Other activities included examination validation, closeout of previously identified inspection findings and a review of selected areas of the licensed operator requalification training program.

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Results:

Operator Pass/Fail:

OVERALL RESULTS	Total No.	No. Passed	% Passed	No. Failed	% Failed
Reactor Operator	7	7	100%	0	0%
Senior Operator	5	5	100%	0	0%

Examiners identified a non-cited violation concerning the failure of some licensed operators to perform control manipulations required by 10 CFR 55.59(c)(3), "On-the-job training," NCV 50-325/94-300-01 (paragraph 2.e).

Examiners identified a weakness in plant procedures and labeling (paragraph 2.f).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*H. Beane, Quality Control Manager
K. Bowden, Licensed Operator Initial Training Instructor
*M. Braley, Nuclear Assessment Department Manager
*J. Cowan, General Manager, Unit 1
*B. Geise, Simulator Manger
E. Hawkins, Simulator Engineer
*D. Hicks, Training Manager
*G. Honma, Licensing Manager
*W. Levis, Regulatory Affairs Manager
H. McDaniel, Licensed Operator Requalification Instructor
*C. Robertson, Environmental and Radiological Controls Manager
*J. Titrington, Operations Manager, Unit 2
*M. Williams, Operator Training Manger

Other licensee employees contacted included instructors, engineers, technicians, operators, and office personnel.

NRC Personnel

*R. Prevatte, Senior Resident Inspector - Brunswick

*Attended exit interview

Acronyms and initialisms used in this report are listed in the last paragraph.

2. Discussion

a. Scope

Examiners administered initial written examinations and operating tests to seven RO and five SRO candidates at the Brunswick Steam Electric Plant during the period January 24-28, 1994. Activities included examination development, review and administration; and assessment of candidate performance. Also conducted was a review of results of the licensee's audit of control manipulations, an assessment of selected procedures and labels, and an assessment of simulator fidelity.

b. Examination Development

Facility representatives pre-reviewed the written examination and operating tests under security agreement to assure an accurate, operationally oriented examination. They conducted this review January 10-14, 1994. The licensee normally conducts the written examination validation and review activities in the NRC Region II offices. Due to the Christmas holiday, however, the licensee reviewed all examination materials at the Brunswick training center. The licensee had no post-examination comments.

c. Examination Administration

The NRC conducted the examination during the week of January 24, 1994. The team conducted simulator and walkthrough tests January 24-27 and the written examination on January 28, 1994. The simulator performed well and supported the examination without causing delays. Brunswick training department personnel assisted with the administration of the examinations by operating the simulator and providing current plant procedures to support JPM administration. This assistance contributed greatly to the validity of the examinations.

d. Candidate Performance

The candidates demonstrated generally good performance during walkthrough and simulator tests. The candidates approached tasks assigned during the simulator examination in a cautious and deliberate manner. Although some candidates exhibited individual weaknesses in some areas, this indicated no generic weaknesses. Though their overall performance was adequate, the candidates exhibited more anxiety than normal.

e. On-the-job Training Requirements

The training department conducted an audit of their records to determine if all operators had completed plant control manipulations required by 10 CFR 55.59(c)(3), "On-the-job training." The audit identified 12 operators who had not performed all of the required manipulations. This occurred because the training department management had not assigned responsibility for reviewing the records for compliance to anyone in the training department. When the training department detected the error, they notified the affected operators and provided them with the simulator time necessary to complete the control manipulations. The licensee subsequently assigned responsibility for tracking operator's performance of the required manipulations to an individual in the training department. Since the licensee self-identified this violation and took corrective actions and actions to prevent recurrence expeditiously, this violation is identified as Non-cited Violation, NCV 50-325/94-300-01.

f. Plant Procedure and Labeling Deficiencies

The examiners noted problems with plant procedures which had the potential to cause operator errors.

The team used a JPM developed by the Brunswick training department to evaluate candidate's abilities to install electrical jumpers as directed by EOP-01-SEP-01, "Primary Containment Venting." The task evaluated was the installation of jumpers per section 1, steps 4 and 5 of the procedure to allow operation of purge valves and fans. The licensee recently installed a plant modification so that instead of four jumpers, only three were necessary to complete the task. Contrary to this modification, the licensee failed to remove the banana clips on terminals 23 and 35, terminal board B, in cabinet XU-56 associated with the fourth jumper. The NRC performed a check of the EOP locker which revealed that Unit 1 had four jumpers available for this task while Unit 2 had the necessary three jumpers available.

The NRC prepared a JPM to evaluate the candidates' ability to perform standby gas treatment system local deluge system manual operation per section 8.6.2 of OP-10, Standby Gas Treatment System Operating Procedure. Two trains of deluge are required to be activated to properly complete the procedure. Each train must be manually initiated by manually overriding a solenoid valve. The wording of the procedure misleads an operator into only initiating one train of deluge. An unknown person had written "manual initiation" in temporary marker on the casing of one of the solenoid valves that is overridden for manual initiation. The other solenoid valve was not similarly marked. The combination of a poorly worded procedure and the presence of unauthorized, incomplete operator aids prevented some of the candidates from correctly performing this task.

The NRC used a JPM developed by the Brunswick training department to evaluate the candidate's ability to maximize flow from the control rod drive system per EOP-01-SEP-09, CRD Flow Maximization. This task involved local alignment of the CRD system for water level control. The licensee identified valves required to be manipulated by this procedure with steel tags embossed with the valve name and other identifiers. They had hung these tags on the valve using small chains. The lettering on the tags and the location of the valves made it difficult for the candidates to read some of the tags. The design had located many valves inside contaminated areas. This required the operators to take radiological precautions when looking at the tags and operating the valves. Since operators perform emergency procedures under duress, possibly without the benefit of normal lighting, labeling and/or operator aids that allow expeditious location of equipment are essential to the operators.

The examiners also noted that a procedure reference to a plant label was not correct. "Alternate Safe Shutdown Procedure," 0-ASSD-02, Section G, step 9.(c).6 referenced breaker AI2. The correct reference for the breaker was A12.

The team identified the above deficiencies in plant procedures and labeling as a weakness.

g. Simulator Facility

The examiners reviewed the schedule for simulator usage to determine priorities given to the various groups for simulator time. Licensed operator requalification training received first priority, using the day shift. Initial licensed operator training received second priority, typically using the afternoon shift. All other functions, such as simulator maintenance and manager training, received third priority.

3. Action on Previous Inspection Findings

(Closed) IFI 50-324/93-55-02 and 50-324/93-55-02, "Additional Training on Procedures and NRC Information Notice on the Reactor Vessel Water Level Reference Leg Backfill Modification." The inspector reviewed the lesson plan used to provide training to licensed and non-licensed operators on the newly installed reference leg fill system. The inspector Also reviewed class attendance rosters that documented classes during which the material was taught as well as reviewing documentation of directed self-study of the material. The inspector determined that all licensed and non-licensed operators, except one who was out of the country, have received the training. The remaining operator will receive the training when he/she returns. Examiners considered the licensee's corrective actions to be adequate. This Inspector Follow-Up Item is closed.

4. Exit Interview

At the conclusion of the site visit, the examiners met with the plant representatives listed in paragraph 1 to discuss the results of the examination and inspection findings. The licensee did not identify as proprietary any material provided to, or reviewed by the examiners. Dissenting comments were not received from the licensee.

Item Number	Status	Description and Reference
NCV 50-325/94-300-01		Failure of some operators to perform control manipulations required by 10 CFR 55.59(c)(3)
IFI 50-325/93-55-02 IFI 50-324/93-55-02	Closed	Additional training on procedures and NRC information notice on the reactor vessel water level reference leg backfill modification

5. List of Acronyms and Initialisms

CRD	Control Rod Drive
EOP	Emergency Operating Procedure
IFI	Inspector Follow-up Item
JPM	Job Performance Measure
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
OP	Operating Procedure
RO	Reactor Operator
SRO	Senior Reactor Operator

ENCLOSURE 2

SIMULATOR FACILITY REPORT

Facility Licensee: Brunswick 1 & 2

Facility Docket Nos.: 50-325 and 50-324

Operating Tests Administered On: January 24-27, 1994

This form is to be used only to report observations. These observations do not constitute, in and of themselves, 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 that may be used in future evaluations. No licensee action is required solely in response to these observations.

While conducting the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION

There were no findings related to simulator fidelity.

Brunswick 94-300
Master

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 2

CANDIDATE'S NAME: MASTER

FACILITY: Brunswick 1 & 2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 94/01/24

INSTRUCTIONS TO CANDIDATE:

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

TEST VALUE	CANDIDATE'S SCORE	%	
100.00		%	TOTALS
	FINAL GRADE		

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

Candidate's Signature

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

MULTIPLE CHOICE

001	a	b	c	d	___
002	a	b	c	d	___
003	a	b	c	d	___
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042	a	b	c	d	___
043	a	b	c	d	___
044	a	b	c	d	___
045	a	b	c	d	___

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

046	a	b	c	d	____	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	____
094	a	b	c	d	____
095	a	b	c	d	____
096	a	b	c	d	____
097	a	b	c	d	____
098	a	b	c	d	____
099	a	b	c	d	____
100	a	b	c	d	____

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

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

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

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

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

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

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

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

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

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

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

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

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QUESTION: 001 (1.00)

A reactor scram has occurred. The equalizing valves in the Control Rod Drive Hydraulic System have failed to open. WHICH ONE (1) of the following is the consequence of the failure?

a. Control rod movement could cause excessive control rod speed.

b. Control rods cannot be moved.

c. The scram discharge volume could be pressurized to reactor pressure.

d. The drive water header pressure could exceed 300 psi.

QUESTION: 002 (1.00)

While operating the plant at 30% power the DC power is lost to MSIV 1B21-FO28C. WHICH ONE (1) of the following is the effect on the MSIV?

a. The slow closure capability of the valve will be lost and the valve will maintain position.

b. The slow closure capability of the valve will be lost and the valve will close.

c. The position indication will be lost and the valve will maintain position.

d. The position indication will be lost and the valve will close.

QUESTION: 003 (1.00)

The following conditions exist for unit 1.

The Mode Switch is in RUN
Reactor power 10%
MSIV 1B21-FO22C is stuck at 25% open.

WHICH ONE (1) of the following MSIVs, if closed, will cause a Half Scram signal?

a. 1B21-FO22D

b. 1B21-FO28C

c. 1B21-FO22B

d. 1B21-FO28B

QUESTION: 004 (1.00)

When reactor vessel water level is less than 182" AND reactor feed pump flow is less than 20% rated flow for either pump, the recirculation flow control speed limiter #2 (45% demand) is in effect. WHICH ONE (1) of the following is the purpose of the speed limiter #2?

a. To prevent running the recirculation pump at high speed with the discharge valve only partially open.

b. To provide NPSH protection for the recirculation jet pumps at low feedwater flow.

c. To limit reactor power so that the feedwater system will be able to maintain or recover reactor water level on loss of a reactor feed pump.

d. To limit recirculation flow such that possible operation in the areas of instability is reduced.

QUESTION: 005 (1.00)

WHICH ONE (1) of the following describes how the core flow signal used for the recirculation system indication is generated?

a. The sum of the recirculation pump demand signal and the feedwater demand signal.

b. The sum of the individual recirculation loop flows plus feedwater flow.

c. Five times the "calibrated" jet pump flows.

d. The sum of all individual jet pump flows.

QUESTION: 006 (1.00)

Concerning the Flow Control Operational Map (attached). WHICH ONE (1) of the following describes why the "Minimum Recirculation Pump Speed" line has been established for operation with one recirculation pump?

a. Operation above the minimum recirculation pump speed line assures the reduced EOC-RPT MCPR limits are valid.

b. Operation above the minimum recirculation pump speed line assures adequate core flow to assure the flux distribution assumed in the accident analysis.

c. Operation above the minimum recirculation pump speed line assures enough coastdown flow following a loss of pumps

d. Operation above the minimum recirculation pump speed line assures sufficient flow to minimize reactor vessel bottom head temperature gradients.

QUESTION: 007 (1.00)

The following conditions exist for the RWCU system:

RPV water level	133 inches
RWCU differential flow	35 gpm
RWCU room temperature	135 deg. F
RWCU system flow	80 gpm
RWCU pump cooling water	145 deg. F
All other parameters are normal	

WHICH ONE (1) of the following DIRECT automatic actions will occur?

a. The RWCU pump will trip.

b. A Group three isolation will occur.

c. The RWCU pump will trip and a group three isolation will occur.

d. The F004 valve closes.

QUESTION: 008 (1.00)

The Standby Gas Treatment System is in standby. WHICH ONE (1) of the following systems can fail and leave the SBGT in a condition in which it will automatically start and perform its intended function?

a. Process Radiation Monitoring System

b. 480 Volt AC System

c. Reactor Building Ventilation System

d. Interruptible Air Systems

09 (1.00)

oup 8 isolation has occurred. Other plant parameters are as

eam dome pressure is 125 psig.
essure is 2.1 psig.
ilding exhaust Rad. is 2 mr/hr.
ilding exhaust temperature is 90 deg. F.

(1) of the following groups of components HAVE ISOLATED?

st Accident Sampling System valves AND HPCI Turbine Exhaust
cuum Breaker (E41-FO75)

ntainment Atmospheric Control System valves AND RNA-SV-5261
d RNA-SV-5262, Div. 1 and Div. 2 RX Building Non-Interruptible
r to Drywell Isolation Valve

PCI Turbine Exhaust Vacuum Breaker (E41-FO75) AND RCIC Exhaust
ine Vacuum Breaker (E51-F062)

CIC Exhaust Line Vacuum Breaker (E51-F062) AND RNA-SV-5261
nd RNA-SV-5262, Div. 1 RX Building Non-Interruptible Air to
rywell Isolation Valve

QUESTION: 009 (1.00)

A valid Group 8 isolation has occurred. Other plant parameters are as follows:

Reactor steam dome pressure is 125 psig.
Drywell pressure is 2.1 psig.
Reactor building exhaust Rad. is 2 mr/hr.
Reactor building exhaust temperature is 90 deg. F.

WHICH ONE (1) of the following groups of components HAVE ISOLATED?

a. Post Accident Sampling System valves AND HPCI Turbine Exhaust Vacuum Breaker (E41-FO75)

b. Containment Atmospheric Control System valves AND RNA-SV-5261 and RNA-SV-5262, Div. 1 and Div. 2 RX Building Non-Interruptible Air to Drywell Isolation Valve

c. HPCI Turbine Exhaust Vacuum Breaker (E41-FO75) AND RCIC Exhaust Line Vacuum Breaker (E51-F062)

d. RCIC Exhaust Line Vacuum Breaker (E51-F062) AND RNA-SV-5261 and RNA-SV-5262, Div. 1 RX Building Non-Interruptible Air to Drywell Isolation Valve

QUESTION: 010 (1.00)

Assuming the level in the suppression pool remains above the elevation of the downcomer openings, WHICH ONE (1) of the following is the Heat Capacity Level Limit?

a. The highest suppression pool level after RPV depressurization that will NOT cover the downcomer vacuum breakers.

b. The lowest suppression pool level after RPV depressurization that will NOT cover the downcomer vacuum breakers.

c. The highest suppression pool water level at which initiation of RPV depressurization will NOT result in exceeding the Heat Capacity Temperature Limit.

d. The lowest suppression pool water level at which initiation of RPV depressurization will NOT result in exceeding the Heat Capacity Temperature Limit.

QUESTION: 011 (1.00)

WHICH ONE (1) of the following describes the operation of the Standby Liquid Control (SLC) pumps and/or squib valves?

a. When the SLC pumps are started from the RTGB control switch, depending on switch position, one pump will start and one squib valve will fire.

b. When the SLC pumps are started from the RTGB control switch, depending on switch position, one pump will start and both squib valves will fire.

c. When the SLC system is initiated from the RTGB, the pumps will only start if the local control switches are in the AUTO position and both squib valves will fire.

d. When the SLC pumps are started at the local pushbutton station the squib valves will fire and the RWCU will remain unaffected.

QUESTION: 012 (1.00)

During a transient RPV pressure has increased to 1120 psig and started to fall due to the opening of SRVs. RPV level has remained in the normal operating range. Assume no operator action and the SRVs opened at their setpoint.

WHICH ONE (1) of the following describes the number of OPEN SRVs?

a. Four

b. Seven

c. Eight

d. Eleven

QUESTION: 013 (1.00)

All the required conditions have been met for ADS initiation and depressurization is in progress.

WHICH ONE (1) of the following describes how the Automatic Depressurization System (ADS) will be affected if all the RHR and Core Spray Pumps trip off?

a. Depressurization stops; manual initiation using pushbuttons is required to re-establish depressurization when a Core Spray or RHR pump is restored.

b. Depressurization stops; automatic depressurization will be reestablished when a core spray or both pumps in a RHR loop are restored.

c. Depressurization continues without pumps running.

d. Depressurization continues for 105 sec and if no RHR or core spray pumps are back in service, then depressurization stops.

QUESTION: 014 (1.00)

The reactor is operating when a transient occurs that automatically initiates the RCIC system. Following the initiation an isolation signal automatically isolates the RCIC system. No manual actions have been taken on the RCIC system except the operator resets the isolation logic when the isolation signal clears and resets the trip and throttle valve. WHICH ONE (1) of the following is the expected system response providing the initiating signal exists?

a. The system will reisolate on a "RCIC Turbine Exhaust Diaphragm High Pressure" signal.

b. The RCIC turbine will trip on overspeed.

c. Water hammer will occur in the RCIC pump discharge line.

d. The RCIC turbine will remain shut down.

QUESTION: 015 (1.00)

You are performing a test on the RCIC isolation logic when you place the RCIC Logic A selector switch to the TEST position. WHICH ONE (1) of the following signals will be bypassed for that division?

a. RCIC Equipment Room High Differential Temperature

b. RCIC Turbine Exhaust Diaphragm High Pressure

c. RCIC Steam line Low Flow

d. RCIC Low Pump Suction Pressure

QUESTION: 016 (1.00)

WHICH ONE (1) of the following HPCI turbine trips is required to be manually reset by the operator. Assume in each choice that the HPCI turbine was tripped from the stated parameter and all parameters are now in the normal operating range.

a. Reactor High water Level

b. HPCI Pump Low Suction Pressure

c. Turbine High Exhaust Pressure

d. Overspeed

QUESTION: 017 (1.00)

WHICH ONE (1) of the following pressures is the point at which the Residual Heat Removal System operating in the Low Pressure Coolant Injection (LPCI) mode will BEGIN to inject water into the reactor recirculation system as reactor pressure lowers?

a. 425 psig

b. 395 psig

c. 195 psig

d. 115 psig

QUESTION: 018 (1.00)

The following plant conditions exist:

Reactor pressure	900 psig
Reactor level	185 inches

WHICH ONE (1) of the following describes the LPCI injection valves that can be opened with NO LPCI injection signal present?

a. F015A ONLY

b. F017A ONLY

c. BOTH F015A AND F017A at the same time

d. EITHER F015A OR F017A but not at the same time

QUESTION: 019 (1.00)

During reactor operation a small break LOCA has occurred and drywell pressure is increasing. WHICH ONE (1) of the following is the minimum drywell pressure that the containment spray valves permissive allows opening E11-F016 and E11-F021, drywell spray valves.

a. 2.0 psig.

b. 2.5 psig.

c. 2.7 psig.

d. 2.9 psig.

QUESTION: 020 (1.00)

The plant is in normal operation when a break occurs in the core spray piping inside the reactor vessel between the core shroud and the vessel wall. WHICH ONE (1) of the following describes the behavior of the core spray sparger line break detection?

The core plate/core spray sparger differential pressure indication will:

a. move from negative pressure towards more negative pressure.

b. move from positive pressure towards more positive pressure.

c. move from positive pressure towards negative pressure.

d. move from negative pressure towards positive pressure.

QUESTION: 021 (1.00)

The core spray system has initiated on a valid signal. It is running normally when there is a loss of AC power to pump 1A. This results in the pump tripping. WHICH ONE (1) of the following is the response of the pump when AC power is restored?

a. The pump must be manually restarted.

b. The pump will automatically restart immediately.

c. The pump will automatically restart after a 10 second time delay.

d. The pump will automatically restart after a 15 second time delay.

QUESTION: 022 (1.00)

During a reactor start up, all SRM detectors may be fully withdrawn under WHICH ONE (1) of the following conditions?

a. When all SRM detectors are between 100 and 50,000 cps.

b. When all SRM detectors are between 50,000 and 100,000 cps.

c. When 4 IRM range switches are above range 4.

d. When all operable IRM range switches are above range 3.

QUESTION: 023 (1.00)

During a reactor start up you find indication that the SRM system has lost monitoring capability and you receive a rod block. The SRM drive control circuits are still operable. WHICH ONE (1) of the following power sources has been lost?

a. 24 Volt DC

b. 120 Volt AC

c. RPS bus

d. UPS

QUESTION: 024 (1.00)

During reactor power ascension the IRM reading for one channel is:

Numerical value 20
Range 5
Red Scale range (0-40)

You have been directed to range up this channel to Range 6. WHICH ONE (1) of the following is the expected numerical value on Range 6, (0-125)?

a. 20

b. 51

c. 60

d. 62

QUESTION: 025 (1.00)

Unit 2 is being shutdown by rod insertion. IRM "B" is ranged down and the reading on the black scale increases to 75/125. WHICH ONE (1) of the following is the correct plant response for these conditions?

a. Half scram in RPS B

b. Half scram in RPS A

c. Rod select block

d. Rod withdrawal block

QUESTION: 026 (1.00)

WHICH ONE (1) of the following is the protective action when the reactor mode switch is in STARTUP and there are fewer than 11 LPRM detectors in OPERATE for APRM channel "C"? The shorting links are installed.

a. Rod select block

b. Half SCRAM on RPS A

c. Half SCRAM on RPS B

d. Full SCRAM

QUESTION: 027 (1.00)

WHICH ONE(1) of the following will cause a Rod Block Monitor DOWNSCALE trip?

a. 40% of the LPRM inputs available

b. No rod selected

c. Mode switch out of operate

d. 94% of selected reference power

QUESTION: 028 (1.00)

WHICH ONE (1) of the following statements describes the response of the TIP system if conditions indicating a Loss of Coolant Accident (LOCA) occur when a TIP detector is inserted beyond the ball valve and the TIP machine power is turned off?

a. The TIP logic is defeated in this condition and a Group 2 isolation signal will not occur on this TIP probe.

b. The ball valve closes regardless of the position of the TIP, shearing the probe and isolating the TIP system.

c. The ball valve closes only after the TIP system transfers to manual reverse mode of operation and withdraws into the shield.

d. The TIP system transfers to reverse and fast speed regardless of its position, then withdraws into the indexer.

QUESTION: 029 (1.00)

During reactor operation condensate pump 2A is running. The pump selector switch is in AUTO and the suction valve closes to 50%.

WHICH ONE (1) of the following is the response of the condensate pump?

a. The pump will trip due to low suction pressure and remain tripped.

b. The pump will trip due to low suction pressure and AUTO restarts when conditions return to normal.

c. The pump will trip due to the 50% closure of the suction valve and remain tripped.

d. The pump will remain running.

QUESTION: 030 (1.00)

WHICH ONE (1) of the following will cause the reactor feed pump (RFP) recirculation valve to AUTO OPEN during a RFP start up?

a. RFP suction pressure 40 psig and MSC logic has NOT sealed in.

b. Pump flow is less than 2500 gpm and RFP suction pressure 45 psig.

c. Pump flow is less than 2800 gpm and MSC logic has sealed in.

d. RFP suction valve open and MSC logic has NOT sealed in.

QUESTION: 031 (1.00)

Unit 2 is operating at 100% power with the FWLCS in three element control. WHICH ONE (1) of the following failures would cause a reduction of NPSH to the recirculation pumps?

a. Loss of one feed flow input to FWLCS

b. The selected level instrument fails low

c. Loss of all feed flow inputs to FWLCS

d. One steam flow signal fails low

QUESTION: 032 (1.00)

The 30 minute holdup in the Off-Gas subsystem collects noncondensible gasses from WHICH ONE (1) of the following sources?

a. Steam jet air ejectors

b. Mechanical vacuum pump

c. Steam packing exhauster

d. Moisture separator reheater vents

QUESTION: 033 (1.00)

WHICH ONE (1) of the following predicts the automatic response of the Augmented Off-Gas System? Assume each parameter (one-at-a-time) is above the trip setpoint.

	H2 Downstream of Recombiner (High)	Moisture Downstream of Guard Bed (High)	System Flow (Hi-Hi)
a.	Bypass and Isolate	Bypass and Isolate	Bypass only
b.	Bypass only	Bypass and Isolate	Bypass only
c.	Bypass and Isolate	Bypass only	Bypass and Isolate
d.	Bypass only	Bypass only	Bypass and Isolate

QUESTION: 034 (1.00)

When monitoring for minor fuel element defects during planned control rod movements prior to reactor shutdown, WHICH ONE (1) of the following monitoring systems is used?

a. Flux Tilt Monitor System

b. Liquid Process Radiation Monitoring System

c. NUMAC Display and Control System

d. RM-23 Display and Control System

QUESTION: 035 (1.00)

WHICH ONE (1) of the following power sources supplies the main stack radiation monitors, recorders and sample skids?

a. RPS distribution panel

b. Vital 120 VAC distribution panel 2AB-RX

c. Unit 2, 120V UPS distribution panel 2A

d. Vital 480 VAC panel MCC-2XG

QUESTION: 036 (1.00)

Given the conditions below:

- Reactor water level +167"
- Drywell pressure 1.6 psig
- HI HI Rad condition for Radwaste Effluent discharge exist

Other annunciators in alarm are:

- DRYWELL FLR DRN SUMP LVL HI
- DRYWELL FLR DR SUMP LEAK HI

Concerning the Radwaste and Floor Drain system, WHICH ONE (1) of the following actions should have occurred?

a. Radwaste Discharge valves D12-V27A and D12-V27B have opened and both drywell floor drain sump pumps have started.

b. The lead drywell floor drain pump will stop and the standby drywell floor drain pump starts.

c. Radwaste Discharge valves D12-V27A and D12-V27B close.

d. The lead and standby drywell floor drain pumps stop.

QUESTION: 037 (1.00)

During refueling a rod was moved to position 08. Attempts to move a second rod resulted in a rod block. After investigating, the operator found the refueling platform over the spent fuel storage area with its hoist loaded and the fuel grapple down. WHICH ONE (1) of the following has ALL the mode switch positions where this interlock is active?

a. Refuel

b. Startup

c. Startup and Refuel

d. Startup, Refuel and Run

QUESTION: 038 (1.00)

During refueling the refueling SRO recognized that the frame mounted hoist "LIFT" function deenergized. WHICH ONE (1) of the following can cause this event to occur? Rod 22-31 is at position 02 and the mode switch is in REFUEL.

a. A second control rod is withdrawn.

b. The main grapple is loaded with fuel over the core.

c. The mode switch is repositioned to STARTUP.

d. The frame mounted hoist is loaded with fuel near the core.

QUESTION: 039 (1.00)

During normal operation the fire system pressure dropped to 93 psig and then increased to 126 psig. WHICH ONE (1) of the following describes the fire system pump status if there NO operator action taken?

a. Jockey pumps are running, diesel driven fire pump is running and the motor driven fire pump is running.

b. Jockey pumps are off, diesel driven fire pump is off and the motor driven fire pump is running.

c. Jockey pumps are running, diesel driven fire pump is off and the motor driven fire pump is running.

d. Jockey pumps are off, diesel driven fire pump is running and the motor driven fire pump is off.

QUESTION: 040 (1.00)

WHICH ONE (1) of the following describes the actions that occur when the control room HVAC system detects a high chlorine condition during normal operations?

a. The control room remains in the normal mode except the normal supply and exhaust are isolated.

b. The control room goes into a recirculation mode and only the normal supply path isolates.

c. The control room goes into a recirculation mode bypassing the HEPA filters and the cable spreading room ventilation dampers isolate and the mechanical equipment ventilation damper isolates.

d. The control room remains in the normal mode, supply and exhaust paths isolate along with the mechanical equipment room dampers.

QUESTION: 041 (1.00)

While a control rod is being inserted using the EMERGENCY IN position of the ROD OUT NOTCH OVERRIDE SWITCH, rod motion stops. WHICH ONE (1) of the following could have terminated rod insertion?

a. the automatic sequence timer deenergizes

b. loss of power to the "withdraw" Directional Control Valve

c. a RWM select block

d. a RWM insert block

QUESTION: 042 (1.00)

WHICH ONE (1) of the following describes the MINIMUM parameter(s) for which the Rod Worth Minimizer System is completely and automatically BYPASSED?

a. Steam flow and feed flow greater than 35%.

b. Steam flow greater than 20%.

c. Steam flow greater than 35%.

d. Steam flow and feed flow greater than 20%.

QUESTION: 043 (1.00)

The reactor is shutdown with IRMs A, F, and G on range 3, the plant experiences an IRM HI-HI condition and a reactor scram signal is generated. WHICH ONE (1) of the following identifies possible mode switch position(s) for this event?

a. Refuel only

b. Startup only

c. Startup and Refuel

d. Refuel, Startup and Run

QUESTION: 044 (1.00)

WHICH ONE (1) OF the following is the source of the SCRAM signal generated for the Turbine Control Valve Fast Closure Trip?

a. Position limit switches on the control valves.

b. Position rate of change monitors on the control valves.

c. EHC regulator servomotor position signal.

d. Pressure switches on the control valve hydraulic oil.

QUESTION: 045 (1.00)

A 4160 VAC breaker has been tested and the clearance removed. The breaker passed the test (charging spring motor toggle to OFF, breaker closed and then opened) and racked in. WHICH ONE (1) of the following is the result of the operator failing to reclose the charging motor toggle switch?

a. The breaker has normal control power indication on the control panel and can be closed and opened one time.

b. The breaker has no control power indication on the control panel and can be closed and opened one time.

c. The breaker has normal control power indication on the control panel but cannot be operated from the control panel.

d. The breaker has no control power indication on the control panel and cannot be operated from the control panel.

QUESTION: 046 (1.00)

WHICH ONE (1) of the following system loads are affected by a loss of power to E-5 480 Volt bus?

a. Diesel Generator #2 Jacket Water Cooling Pump

b. UPS Unit 1 Alternate Feed

c. DG Vent Exhaust Fan B

d. Loop 1A Core Spray Valves

QUESTION: 047 (1.00)

WHICH ONE (1) of the following would be a source of radioactive leakage into the Reactor Building Closed Cooling Water System?

a. Reactor Water Cleanup

b. Control Rod Drive

c. Penetration Cooling

d. Reactor Feedwater

QUESTION: 048 (1.00)

WHICH ONE (1) of the following isolation signals will NOT close RWCU Outboard Isolation Valve (G31-F004)?

a. Low Level #2

b. NRHX High Outlet Temperature

c. High Drywell Pressure

d. SLC Initiation

QUESTION: 049 (1.00)

The Diesel Generators have started in response to a LOCA signal. WHICH ONE (1) of the following signals will shutdown the Diesel Engines?

a. Lube oil pressure - 22 psig

b. Lube oil temperature - 197 deg F

c. Jacket water pressure - 15 psig

d. Jacket water temperature - 187 deg F

QUESTION: 050 (1.00)

With reactor power less than 2% and the mode switch in STARTUP, WHICH ONE (1) of the following inputs to the RPS is bypassed?

a. IRM Hi Hi.

b. Flow biased APRM

c. Main Steam Line Hi Rad.

d. Turbine stop valve closure

QUESTION: 051 (1.00)

The following reactor conditions exist:

Core flow is 37%

Reactor Power is 22%

Recirculation flow is 20%

WHICH ONE (1) of the following is the maximum flow biased APRM SCRAM setpoint required by Technical Specifications?

a. Less than or equal to 55.2%

b. Less than or equal to 77.2%

c. Less than or equal to 78.5%

d. Less than or equal to 90.4%

QUESTION: 052 (1.00)

The reactor is operating with the "A" and "B" CSW pump in service when the CSW header pressure drops to 35 psig. The operator verifies the SWV-3 and SWV-4 are at the throttled position. The "C" CSW pump is INOPERABLE. WHICH ONE (1) of the following actions should be taken?

a. Trip the "A" and "B" CSW pumps and manually SCRAM the reactor.

b. Reduce all Service Water flows to maintain CSW header pressure above 30 psig.

c. Verify the RBCCW containment isolation valves have isolated and manually SCRAM if drywell pressure increases to 2.0 psig.

d. Monitor the CSW header pressure and manually SCRAM the reactor if pressure falls to 30 psig.

QUESTION: 053 (1.00)

The reactor is operating and the Reactor Building Standby Compressor is in AUTO. As air pressure decreases WHICH ONE (1) of the following will start the Reactor Building Standby Compressor?

a. Service air header pressure less than 111 psig.

b. SCRAM pilot valve air header pressure less than 55 psig.

c. Noninterruptible instrument air header pressure less than 95 psig.

d. Pneumatic Nitrogen system supply pressure less than 95 psig.

QUESTION: 054 (1.00)

The reactor is operating when there is a loss of offsite power. As the reactor operator you have verified the reactor SCRAM. The SRO instructs you to verify the PCIS isolations as required by AOP-36.1. WHICH ONE (1) of the following sets of PCIS isolations should you verify?

a. Group 1, 2, 3, 4 and 6.

b. Group 1, 2, 3, 5 and 6.

c. Group 1, 2, 3, 5 and 8.

d. Group 1, 2, 3, 6 and 8.

QUESTION: 055 (1.00)

While operating the reactor at 90% power the "A" recirculation pump experiences a SPEED CONTROL SIGNAL FAILURE. You have ensured this is not a recirculation pump runback and locked the scoop tube on the "A" recirculation pump. You have verified operation in Region "C" with no thermal hydraulic instabilities. WHICH ONE (1) OF the following is the required immediate operator action?

a. INCREASE core flow to greater than 35 Mlbs/hr using the unaffected recirculation pump.

b. DECREASE core flow to less than 35 mlbs/hr using the unaffected recirculation pump.

c. WITHDRAW control rods to the 100% rod line per the Nuclear Engineer's instructions.

d. Continue to monitor for thermal hydraulic instabilities and obtain approval of Nuclear Engineer to insert control rods.

QUESTION: 056 (1.00)

WHICH ONE (1) of the following conditions will NOT be mitigated by the use of procedure EOP-02, "Primary Containment Control"?

a. Primary containment temperature of 140 deg. F.

b. Suppression pool level of -32 inches.

c. Primary containment pressure of 1.5 psig.

d. Primary containment Hydrogen level of 1.8%.

QUESTION: 057 (1.00)

WHICH ONE (1) of the following conditions is a symptom of Condensate and Feedwater failure per AOP-23.0?

a. Safety relief valves on the LP heaters shell side lifting.

b. Indicated reactor vessel level at +190 inches.

c. Steam flow/feed flow mismatch as indicated on the recorder on Panel P603.

d. High offgas flow as indicated on the RTGB instrumentation.

QUESTION: 058 (1.00)

During reactor operation you notice reactor power and generator power increasing. One recirculation pump current indication is increasing. Reactor power is 80%, total core flow is 65 million lbs/hr and the recirculation pump mismatch is 15%. WHICH ONE (1) of the following is the immediate operator action?

a. Trip the affected recirculation pump.

b. Lock the scoop tube on the affected recirculation pump.

c. Place the affected recirculation pump in MANUAL control.

d. Manually SCRAM the reactor.

QUESTION: 059 (1.00)

WHICH ONE (1) of the following conditions requires a manual reactor SCRAM if a failure of the RBCCW system occurred?

a. RWCU isolation.

b. High temperature alarm on the RBCCW heat exchanger outlet.

c. Drywell chiller trip.

d. RBCCW pressure less than 60 psig.

QUESTION: 060 (1.00)

WHICH ONE (1) of the following conditions requires entry into an EOP?

a. Failing to reenergize the UPS bus.

b. Primary containment temperature at 131 degrees F.

c. Secondary Containment dp is -0.19 inches H_2O.

d. Reactor Building ventilation exhaust radiation level at 2 Mr/hr.

QUESTION: 061 (1.00)

During an ATWS with the reactor at high power, EOP-01-RSP directs that recirculation be run back to 10% prior to tripping the Reactor recirculation pumps. WHICH ONE (1) of the following is the basis for this action?

a. Prevent MSIV closure on high flow.

b. Promote boron mixing.

c. Prevent Main Generator reverse power trip.

d. Prevent Main Turbine high RPV water level trip.

QUESTION: 062 (1.00)

WHICH ONE (1) of the following is an indication of a jet pump failure that may require entry into AOP 04.4, "Jet Pump Failure"?

a. Decrease in indicated total core flow.

b. Core plate differential pressure decrease.

c. Recirculation loop flow decrease in the loop with the failed jet pump.

d. Recirculation pump discharge pressure increase on the loop with the failed jet pump.

QUESTION: 063 (1.00)

The conditions below develop due to a recirculation system leak.

Drywell pressure	6.7 psig
RPV level	+40 inches and decreasing
RPV pressure	405 psig

WHICH ONE (1) of the following explains the status of the CORE SPRAY (CS) system?

a. The CS minimum flow bypass valve (E21-F031A) for loop A is CLOSED and will remain closed until flow increases above 450 gpm.

b. The CS inboard injection valves (E21-F005A and B) should be OPEN and the minimum flow bypass valves (E21-F031A and B) should be CLOSED.

c. The CS inboard injection valves (E21-F005A and B) should be CLOSED and the minimum flow bypass valves (E21-F031A and B) should be OPEN.

d. The CS inboard injection valves (E21-F005A and B) and minimum flow bypass valves (E21-F031A and B) should be OPEN.

QUESTION: 064 (1.00)

Given the following conditions:

Reactor water level +172 inches
Drywell pressure 1.72 psig
Primary containment average temp. 138 F
Reactor Bldg Vent Exhaust Rad Level 3.3 mR/hr
The "Reactor Bldg Vent Rad Hi Exhaust" annunciator has NOT alarmed

WHICH ONE (1) of the following lists of EOPs are you required to enter?

a. EOP-01-RVCP Reactor Vessel Control and EOP-02-PCCP Primary Containment Control.

b. EOP-01-RVCP Reactor Vessel Control and EOP-03-SCCP Secondary Containment Control.

c. EOP-02-PCCP Primary Containment Control and EOP-03-SCCP Secondary Containment Control.

d. EOP-03-SCCP Secondary Containment Control and EOP-04-RRCP Radioactivity Release Control.

QUESTION: 065 (1.00)

During a plant transient, a failure of the RCIC logic bus 'B', due to a ground fault occurs. The RCIC system is manually started. WHICH ONE (1) of the following describes how the RCIC system would respond to RPV water level?

a. It would shut down on high RPV level and would automatically start on a decreasing RPV level.

b. It would NOT shut down on high RPV level and would automatically start on a decreasing RPV level.

c. It would shut down on high RPV level and would NOT automatically start on a decreasing RPV level.

d. It would NOT shut down on high RPV level and would NOT automatically start on a decreasing RPV level.

QUESTION: 066 (1.00)

While operating Unit 2 in a steady state at 78% power the below conditions were observed by the operator:

Total feedwater flow decreased slightly and returned normal.

2B RFP suction flow increased slightly and discharge flow decreased.

Reactor water level decreased slightly and returned to normal.

Reactor power remained relatively constant.

Feedwater level control is in three-element control.

Master level controller is in AUTO

2B RFP controller in AUTO and 2A RFP controller is in MANUAL

WHICH ONE (1) of the following explains the plant response?

a. The selected RPV level transmitter failed and an automatic transfer to the redundant transmitter has occurred.

b. The output signal from 2A RFP controller had decreased slightly an then returned to normal.

c. The control rod pattern has been modified using rods near the outer edges of the core.

d. The 2B RFP minimum flow valve had sensed a low feedwater flow signal and gradually opened and then stayed opened.

QUESTION: 067 (1.00)

During an event which initiated HPCI, WHICH ONE (1) of the following describes how the HPCI system would respond if annunciator A-01 2-5, "HPCI INVRTR POWER FAILURE", went into an alarm condition?

a. HPCI Pump discharge pressure would increase as the control valves open.

b. HPCI steam line drain pot drain valve and condensate discharge valve close.

c. HPCI system logic will sense a low CST level and open the torus suction valves.

d. HPCI discharge flow will decrease and RTGB HPCI flow indication will fail downscale.

QUESTION: 068 (1.00)

While at 75% power the following actions occur:

PCB-29A and PCB-29B generator output breakers trip and are locked out.

UAT Breakers to Bus 2C and 2D are tripped and locked out.

The UAT (alt.) Breaker to bus 2B is locked out.

The Main Turbine tripped.

Stator cooling pumps A and B tripped.

WHICH ONE (1) of the following can be contributed to all the above actions occurring?

a. 230KV bus differential overcurrent.

b. Loss of Substation E8.

c. The SAT lock-out relay has energized.

d. Main Transformer fault pressure.

QUESTION: 069 (1.00)

Concerning the Service Air system, WHICH ONE (1) of the following describes the reason for the service air isolation valves closing at 105 psig?

a. Assure an adequate supply of breathing air throughout the plant.

b. Assure the minimum allowable pressure is maintained on the Service Air system.

c. Assure an adequate supply of air to the RNA System.

d. Assure an air supply to drywell loads under all expected operating conditions.

QUESTION: 070 (1.00)

The reactor is operating at full power. As the Reactor Operator you notice the main condenser vacuum is 26 inches Hg and slowly decreasing. The circulating water pumps are verified to be operating properly. WHICH ONE (1) of the following actions is required by AOP-37.0, "Low Condenser Vacuum"?

a. Reduce generator load, as necessary, to maintain condenser vacuum greater than 25 inches Hg.

b. Increase gland sealing steam pressure to greater than 5 psig.

c. Manually SCRAM the reactor.

d. Start BOTH mechanical vacuum pumps.

QUESTION: 071 (1.00)

The reactor is operating at 25% power when the main condenser vacuum decreases to 22" Hg. WHICH ONE (1) of the following events is the direct action? Assume the Mode Switch is in RUN.

a. Main turbine trips.

b. Turbine Bypass Valves close.

c. Group 1 isolation.

d. Automatic reactor SCRAM.

QUESTION: 072 (1.00)

Unit 1, "A" loop of RHR, is in shutdown cooling. Plant conditions cause the SDC suction valves, E11-F008 and F009, and the RHR injection isolation valve E11-F015A, to close. Which ONE (1) of the following conditions caused the shutdown cooling isolation?

a. Reactor water level +210 inches

b. Reactor water level +162.5 inches

c. Drywell pressure 2.0 psig

d. Reactor steam dome pressure 140 psig

QUESTION: 073 (1.00)

Unit 2 is conducting an initial start up after refueling and has the RPS shorting links removed. During a tagging evolution, power was removed from SRM B. Which ONE (1) of the following is the plant response?

a. RPS A half scram and Rod block

b. RPS B half scram and Rod block

c. Full scram

d. Rod block

QUESTION: 074 (1.00)

WHICH ONE (1) of the following is the cause of the reactor power decrease when lowering reactor water level during an ATWS?

a. Reduces natural circulation and increases void fraction.

b. Allows fluid density of water inside and outside the shroud to equalize and equally distribute voiding throughout the core.

c. Allows an increase in moderator temperature to promote neutron thermalization.

d. Promotes evaporative cooling and subsequent Doppler broadening.

QUESTION: 075 (1.00)

WHICH ONE (1) of the following sets of plant parameters will directly cause the Main Turbine to trip?

a. Reactor Water Level 191 inches
Turbine Speed 1902 rpm
Condenser Vacuum 25 in Hg
Bearing Oil Supply Header 15 psig

b. Reactor Water Level 210 inches
Turbine Speed 1340 rpm
Condenser Vacuum 24 in Hg
Bearing Oil Supply Header 11 psig

c. Reactor Water Level 161 inches
Turbine Speed 1875 rpm
Condenser Vacuum 26 in Hg
Bearing Oil Supply Header 13 psig

d. Reactor Water Level 155 inches
Turbine Speed 1545 rpm
Condenser Vacuum 23 in Hg
Bearing Oil Supply Header 9 psig

QUESTION: 076 (1.00)

Which ONE (1) of the following methods for alternate rod insertion requires the scram to be reset?

a. Venting the scram air header.

b. Inserting control rods from the Reactor Manual Control System.

c. Venting the over-piston area of the control rod.

d. Using the individual scram test switches.

QUESTION: 077 (1.00)

The Shift Supervisor has determined that a control room evacuation will be required. WHICH ONE (1) of the following is NOT an immediate operator action?

a. Trip the main turbine.

b. Trip both reactor recirculation pumps.

c. Reduce reactor pressure to 700 psig.

d. Place all MSIV switches to AUTO.

QUESTION: 078 (1.00)

Which ONE (1) of the following methods of alternate shutdown cooling should be used if there is an isolation of shutdown cooling and reactor coolant cannot be maintained below 212 deg F?

a. Align TBCCW to supply coolant flow to the RBCCW heat exchangers to supply cooling to the shutdown cooling system.

b. Align service water supply coolant flow to the RBCCW heat exchangers and maximize flow to the drywell and shutdown cooling system.

c. Establish gravity drain from the reactor vessel through the RHR heat exchanger into the torus and use RHR suction from the torus to pump back into reactor vessel.

d. Open one SRV and raise reactor level to establish a flowpath through the SRV to the torus and use RHR suction from the torus to pump back into the reactor vessel.

QUESTION: 079 (1.00)

A reverse power trip is NOT the preferred means to take the main turbine off-line. This is because allowing an automatic reverse power trip of the main turbine:

a. will result in excessive arcing in the main generator output breakers.

b. may cause a pressure spike sufficient to rupture the LP turbine relief diaphragms.

c. will result in an automatic cold start of that unit's diesel generators.

d. may place an unnecessary load on the main turbine thrust bearing.

QUESTION: 080 (1.00)

WHICH ONE (1) of the following indications is used to monitor plant heatup/cooldown rate while in Alternate Shutdown Cooling?

a. RBCCW heat exchanger outlet temperature.

b. RHR heat exchanger outlet temperature.

c. Safety relief valve tailpipe temperature.

d. The running ECCS pump local suction temperature.

QUESTION: 081 (1.00)

Yesterday the reactor was shutdown for refueling and reported subcritical at 10:00 am. It is now 7:00 am and refueling is scheduled to begin. Prior to commencement of refueling operations, SRMs A and B are reading 2 cps and SRMs C and D are reading 8 cps. The level in the fuel pool is 21 feet 2 inches above the fuel assemblies seated in the spent fuel storage rack.

WHICH ONE (1) of the following would prevent removal of spent fuel from the reactor core?

a. The level of water above the fuel assemblies when seated in the spent fuel storage rack is not high enough.

b. The required number of SRM channels are not available per Technical Specifications.

c. The required number of SRM channel trip systems are not available per Technical Specifications.

d. The required time before spent fuel can be removed from the reactor core has not elapsed.

QUESTION: 082 (1.00)

During accident conditions you would like to use the narrow range level instruments to determine RPV level. WHICH ONE (1) of the following sets of parameters are required to determine if the instruments may be used?

a. Drywell temperature and reactor pressure only

b. Indicated level and drywell temperature only

c. Reactor pressure and drywell temperature only

d. Reactor pressure, drywell temperature and indicated level

QUESTION: 083 (1.00)

A fire in the control room of Unit 1 has caused an evacuation of the control room. WHICH ONE (1) of the following actions prevents spurious operation of the SRVs?

a. The supply breakers to 125V DC distribution panels 3A and 3B located on 125/250 switchboards 1A and 1B are opened.

b. DC switchboards 2B/1B loads are stripped except for distribution panels 2B/1B and MCCs 2XDB/1XDB.

c. Battery chargers 1B are transferred to their alternate power supply.

d. De-energize, via keylock switches, the Div. I Nitrogen backup system valves.

QUESTION: 084 (1.00)

During a Site Area Emergency, the AO on building rounds did not report to the Control Room and does not respond to the plant paging system. WHICH ONE (1) of the following is the maximum exposure allowed to an individual in order to search for the unaccounted for operator?

a. Brunswick administrative limits

b. 10CFR20 non-emergency limits

c. 25 REM

d. 75 REM

QUESTION: 085 (1.00)

The Emergency Plan has been activated and a Site Area Emergency declared. All Brunswick management personnel are on site and have assumed their positions. WHICH ONE (1) of the following identifies the two individuals who have the authority to approve entry into a radiation field greater than 100 R/hr?

a. Plant General Manager and Environmental and Radiation Control Manager

b. Plant General Manager and Emergency Response Manager

c. Emergency Response Manager and Site Emergency Coordinator

d. Site Emergency Coordinator and Environmental and Radiation Control Manager

QUESTION: 086 (1.00)

Due to the inoperability of a hand/foot monitor, personnel contamination monitoring is being performed with a pancake GM detector. WHICH ONE (1) of the following states the MINIMUM indications at which individuals shall be considered contaminated?

a. any count rate greater than background

b. count rate of 100 cpm above background

c. count rate of 200 cpm above background

d. count rate of 1000 cpm above background

QUESTION: 087 (1.00)

Entry into WHICH ONE (1) of the following areas requires continuous HP monitoring?

a. Locked High Radiation area

b. Restricted High Radiation area

c. High Contamination area

d. Radioactive Material area

QUESTION: 088 (1.00)

WHICH ONE (1) of the following is indicated by an orange colored handwheel?

a. the handwheel must be turned clockwise to open

b. operation of the valve must be approved by the Shift Supervisor

c. independent verification is required if the valve is operated

d. the valve is a radiological "hot spot"

QUESTION: 089 (1.00)

A level instrument is being returned to service and both the high and low pressure root valves are CLOSED. WHICH ONE (1) of the following states the correct action to perform first to ensure disturbances are not created while returning the instrument to service?

a. CLOSE/check CLOSED the equalizing valve, then open the LOW pressur. ot valve.

b. CLOSE/check CLOSED the equalizing valve, then open the HIGH pressure root valve.

c. OPEN/check OPENED the equalizing valve, then open the LOW pressure root valve.

d. OPEN/check OPENED the equalizing valve, then open the HIGH pressure root valve.

QUESTION: 090 (1.00)

An operator is assigned to work in a room at 90 degrees F WBGT wearing a single cotton blend plus impervious garment. The task will require and occasional ladder climbing and valving to lineup a system and will take approximately two hours to complete. WHICH ONE (1) of the following is the MAXIMUM time the operator may remain in the environment in order to initiate the task and ensure it is brought to a satisfactory and safe stopping point? (See Attached Procedure AI-107)

a. 25 minutes

b. 30 minutes

c. 37.5 minutes

d. 100 minutes

QUESTION: 091 (1.00)

Leads for two phases of a three phase motor (Technical Specification-related equipment) have been momentarily lifted. WHICH ONE (1) of the following states the requirements dictated by this action per procedure AI-59, Jumper and Wire Removal?

a. Jumper wire removal tags must be attached

b. Temporary Modification form must be submitted within 24 hours.

c. Independent verification must be performed per drawings

d. Correct motor rotation must be verified upon retermination

QUESTION: 092 (1.00)

Personnel in the Emergency Organization who do not have a designated emergency assignment when a site evacuation is performed will report to WHICH ONE (1) of the following for instructions/assignments?

a. Control Room Supervisor

b. Operational Support Center Leader

c. Site Emergency Coordinator

d. Immediate Supervisor

QUESTION: 093 (1.00)

WHICH ONE (1) of the following radiation exposure guidelines is the MINIMUM exposure at which the Shift Supervisor may waive the independent verification requirement for a system valve lineup?

a. General area radiation levels are in excess of 10 mr/hr.

b. Ge eral area radiation levels are in excess of 100 mr/hr.

c. Total exposure expected during the independent verification is in excess of 10 mr.

d. Total exposure expected during the independent verification is in excess of 100 mr.

QUESTION: 094 (1.00)

WHICH ONE (1) of the following annunciator window colors designates a setpoint important to reactor safety?

a. Red annunciator window with a red bar.

b. Amber annunciator window with a red bar.

c. Red annunciator window with a blue bar.

d. Amber annunciator window with a blue bar.

QUESTION: 095 (1.00)

10CFR50.54 (x) states that reasonable action that departs from a license condition or technical specification may be taken in an emergency to protect the health and safety of the public. WHICH ONE (1) of the following states the MINIMUM approval required?

a. any licensed reactor operator

b. any licensed senior reactor operator

c. Site Emergency Coordinator

d. General Plant Manager

QUESTION: 096 (1.00)

WHICH ONE (1) of the following states the condition during which entry to the drywell is prohibited?

a. Primary coolant temperature is 250 degrees F.

b. Reactor Pressure Vessel pressure is 250 psig.

c. Primary Containment oxygen concentration is 20%.

d. Reactor power level is 26%.

QUESTION: 097 (1.00)

Annunciator UA-26 4-8, FIRE REACTOR NO 2 EL -17 NE, actuates. Within 30 seconds, eight additional fire alarms in areas adjacent to that area actuate. One fire pump has failed to start. WHICH ONE (1) of the following states the required immediate operator action?

a. Attempt to manually start the failed fire pump within five minutes.

b. Assemble the fire brigade and request support from municipal fire departments within 10 minutes.

c. CLOSE PIV-33, Reactor Building sprinkler system isolation valve within 15 minutes.

d. CLOSE FP-V214, Reactor Building sprinkler system isolation valve within 30 minutes.

QUESTION: 098 (1.00)

Unit 2 is operating at 100% power. WHICH ONE (1) of the following is an entry condition for EOP-4, Radioactive Release Control?

a. Turbine Bldg Vent Rad Monitor, D12-RM-23, indicates 150 uCi/sec.

b. Process off-gas Rad Monitor, D12-RM-601A, indicates 150 mr/hr.

c. Main Steamline Rad Monitor, D12-K603A, indicates 150 mr/hr.

d. A radioactive steam leak in the Turbine Bldg; ventilation INOP.

QUESTION: 099 (1.00)

EOP-2, Primary Containment Control, requires emergency depressurization if suppression pool level cannot be maintained within the safe region of the "SRV Tail Pipe Level Limit" curve. WHICH ONE (1) of the following identifies the plant changes which BOTH drive the plant toward the UNSAFE portion of the curve and an INCREASED possibility of SRV tail pipe failure?

a. decreasing suppression pool water level, decreasing reactor pressure

b. decreasing suppression pool water level, increasing reactor pressure

c. increasing suppression pool water level, decreasing reactor pressure

d. increasing suppression pool water level, increasing reactor pressure

QUESTION: 100 (1.00)

A PCIS Group 1 isolation has resulted in a reactor scram on Unit 2. The plant has been stabilized using CRD for level control and SRVs for pressure control. Current plant conditions are as follows:

RPV level	180 inches
RPV pressure	1000 psig
Reactor power	IRM range 3

"A" Circulating Water Pump running
"A" Condensate Pump running
"A" Condensate Booster Pump running

WHICH ONE (1) of the following conditions would prevent the reopening of MSIVs?

a. Complete loss of 120 VAC.

b. Condenser vacuum is 0 inches.

c. A malfunction prevents resetting the turbine.

d. SJAE "A" first stage steam supply valve will not open.

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

ANSWER: 001 (1.00)

a.

REFERENCE:

SSM 09-B, pg 8, LO 14.b

K/A: 201001A204 [3.8/3.9]

201001A204 ..(KA's)

ANSWER: 002 (1.00)

c.

REFERENCE:

SSM-17-A, pg. 34, LO10.a.

K/A: 239001K201 [2.8/3.2]

239001K201 ..(KA's)

ANSWER: 003 (1.00)

a.

REFERENCE:

SD-25, "Main Steam System", Rev. 12, Page 5

OSM 17-2A, "Main Steam", Rev. 3, Pages 12 - 14, L.O. - 3

KA: 239001K127[4.0/4.1]

239001K127 ..(KA's)

ANSWER: 004 (1.00)

c.

REFERENCE:

SSM 10-2A, pg 22, LO 3.h.

K/A: 295009K301 [3.2/3.3]

295009K301 ..(KA's)

ANSWER: 005 (1.00)

d.

REFERENCE:

SSM 10-2A, pg 19, LO 9

K/A: 202001K101 [3.6/3.7]

202001K101 ..(KA's)

ANSWER: 006 (1.00)

d.

REFERENCE:

SSM 10-2A, pg. 36, LO 23

K/A: 202001K117 [3.1/3.3]

202001K117 ..(KA's)

ANSWER: 007 (1.00)

a.

REFERENCE:

SSM 11-A, pg 11, LO 6

K/A: 204000K401 [2.5/2.5]

204000K401 ..(KA's)

ANSWER: 008 (1.00)

d.

REFERENCE:

LOI-CLS-LP-010-A, pg 28, LO 9.f.&g.&h.&j.

K/A: 261000K601 [2.9/3.1]

261000K601 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

SSM 15-e, pg. 64,27,31,30, LO 9-13

K/A: 223002A102 [3.7/3.7]

223002A102 ..(KA's)

ANSWER: 010 (1.00)

d.

REFERENCE:

LOI-CLS-SM-004-A, pg 30, LO 13.b

K/A: 295030K103 [3.8/4.1]

295030K103 ..(KA's)

ANSWER: 011 (1.00)

b.

REFERENCE:

SML-4 14G, pg 10, LO 11

K/A: 211000A402 [4.2/4.2]

211000A402 ..(KA's)

ANSWER: 012 (1.00)

c.

REFERENCE:

SSM 14-F, pg. 7, LO 3, 5, 6, and 9

K/A: 295007A104 [3.9/4.1]

295007A104 ..(KA's)

ANSWER: 013 (1.00)

b.

REFERENCE:

SSM 14-F, pg. 14, LO 11

K/A: 218000K101 [4.0/4.0]

218000K101 ..(KA's)

ANSWER: 014 (1.00)

d.

REFERENCE:

SML-4 14-C, pg 44, LO 23

K/A: 217000G010 [3.4/3.5]

217000G010 ..(KA's)

ANSWER: 015 (1.00)

a.

REFERENCE:

SML-4 14-C, pg. 43, LO 18

K/A: 217000G007 [3.8/3.7]

217000G007 ..(KA's)

ANSWER: 016 (1.00)

a.

REFERENCE:

SSM 14-2B, pg. 28, LO 4.1 and 22

K/A: 206000K403 [4.2/4.1]

206000K403 ..(KA's)

ANSWER: 017 (1.00)

c.

REFERENCE:

SSM 14D, pg. 20, LO 20

KA: 203000A304 [3.8/3.7]

203000A304 ..(KA's)

ANSWER: 018 (1.00)

d.

REFERENCE:

SSM 14-D, pg 15, LO 9

K/A: 203000K117 [4.0/4.0]

203000K117 ..(KA's)

ANSWER: 019 (1.00)

c.

REFERENCE:

SSM 14-D, pg 15, LO 27

K/A: 230000K403 [3.5/3.6]

230000K403 ..(KA's)

ANSWER: 020 (1.00)

d.

REFERENCE:

SSM 14-E, pg 5, LO 10

K/A: 209001A209 [2.9/3.0]

209001A209 ..(KA's)

ANSWER: 021 (1.00)

d.

REFERENCE:

SSM 14-E, pg. 12, LO 15

K/A: 209001K601 [3.4/3.4]

209001K601 ..(KA's)

ANSWER: 022 (1.00)

d.

REFERENCE:

SSM 25-A, pg. 14, LO 3.a

K/A: 215004K401 [3.7/3.7]

215004K401 ..(KA's)

ANSWER: 023 (1.00)

a.

REFERENCE:

SSM 25-A, pg 16, LO 4

K/A: 215004K602 [3.1/3.3]

215004K602 ..(KA's)

ANSWER: 024 (1.00)

a.

REFERENCE:

SSM 25-B, pg. 6, LO 5

K/A: 215003A401 [3.3/3.3]

215003A401 ..(KA's)

ANSWER: 025 (1.00)

d.

REFERENCE:

SSM252B pg 19

KA: 215003K401 [3.7/3.7]

215003K401 ..(KA's)

ANSWER: 026 (1.00)

b.

REFERENCE:

SSM 25-D, pg 16, LO 3

K/A: 215005K103 [3.4/3.5]

215005K103 ..(KA's)

ANSWER: 027 (1.00)

d.

REFERENCE:

SSM 25-E, pg. 16, LO 3

KA: 215002A203 [3.1/3.3]

215002A203 ..(KA's)

ANSWER: 028 (1.00)

a.

REFERENCE:

SSM 25-F, pg. 20, LO 5.a and b.

KA: 215001K401 [3.4/3.5]

215001K401 ..(KA's)

ANSWER: 029 (1.00)

d.

REFERENCE:

SSM SS17BR6, pg 8, LO 6.a.

K/A: 256000A202 [2.8/2.9]

256000A202 ..(KA's)

ANSWER: 030 (1.00)

c.

REFERENCE:

SML SS172BR6, pg 16 LO 2.g

K/A: 259001K403 [2.7/2.7]

259001K403 ..(KA's)

ANSWER: 031 (1.00)

d.

REFERENCE:

SML SS172, pg. 26, LO 9

K/A: 259002A201 [3.3/3.4]

259002A201 ..(KA's)

ANSWER: 032 (1.00)

a.

REFERENCE:

LOT SS17-2DR3, pg. 12, LO 2 and 3

K/A: 271000G004 [3.4/3.5]

271000G004 ..(KA's)

ANSWER: 033 (1.00)

a.

REFERENCE:

SSM 24-2, pg. 13, LO 5.a., c., e.

K/A: 271000A301 [3.3/3.3]

271000A301 ..(KA's)

ANSWER: 034 (1.00)

a.

REFERENCE:

SSM 03-B, Rev. 1, P 18, LO #1

KA: 272000G007 [3.5/3.5]

272000G007 ..(KA's)

ANSWER: 035 (1.00)

c. [+1.0]

REFERENCE:

SSM 03-B, Rev. 1, P 10, LO #3 , SD-11 Table 3.5

KA: 272000K603 [2.8/3.0]

272000K603 ..(KA's)

ANSWER: 036 (1.00)

c.

REFERENCE:

SSM 16-A, Rev. 0, P 13, LO #11

KA: 268000K106 [2.9/3.2]

268000K106 ..(KA's)

ANSWER: 037 (1.00)

a.

REFERENCE:

SSP29-2AR4, Rev. 4, P. 42 OF 119, LO #6

KA: 234000K402 [3.3/4.1]

234000K402 ..(KA's)

ANSWER: 038 (1.00)

d.

REFERENCE:

SSP29-2AR4, Rev. 4, P. 41 OF 119, LO #6

KA: 234000K502 [3.1/3.7]

234000K502 ..(KA's)

ANSWER: 039 (1.00)

c.

REFERENCE:

SSM 31-A, Rev. 0, P 27, 28 and 29, LO #9

KA: 286000A301 [3.4/3.4]

286000A301 ..(KA's)

ANSWER: 040 (1.00)

c.

REFERENCE:

SSM 35-B, Augmented, P 3, LO #3

KA: 29003A301 [3.3/3.5]

290003A301 ..(KA's)

ANSWER: 041 (1.00)

d.

REFERENCE:

SML 272A, REV AUG, P 19 and 20, LO #9

KA: 201002A402 [3.5/3.5]

201002A402 ..(KA's)

ANSWER: 042 (1.00)

c.

REFERENCE:

SSM 27-B, REV 7, P 4, LO #3

KA: 201002K402 [3.4/3.5]

201002K404 ..(KA's)

ANSWER: 043 (1.00)

c.

REFERENCE:

OTM SS282AR3, P 43 OF 65, LO #8

KA: 212000K412 [3.9/4.1]

212000K412 ..(KA's)

ANSWER: 044 (1.00)

d.

REFERENCE:

OTM SS282AR3, P. 27 OF 65, LO #5

KA: 212000A215 [3.7/3.8]

212000A215 ..(KA's)

ANSWER: 045 (1.00)

c.

REFERENCE:

ISU: SSM 20I, p. 10, LO 3C.

KA: 262001A401 [3.4/3.7]

262001A401 ..(KA's)

ANSWER: 046 (1.00)

d.

REFERENCE:

SSM 20I, TABLE III, LO 8E.

KA: 262001K301 [3.5/3.7]

262001K301 ..(KA's)

ANSWER: 047 (1.00)

a.

REFERENCE:

SSM 12-A, p. 15, LO 4

KA: 204000K104 [2.9/2.9]

204000K104 ..(KA's)

ANSWER: 048 (1.00)

c.

REFERENCE:

SSM 15-e, p. 12, LO 6.

KA: 204000K404 [3.5/3.6]

204000K404 ..(KA's)

ANSWER: 049 (1.00)

a.

REFERENCE:

SSM 20-D, p. 29, LO 4a.

KA: 264000K402 [4.0/4.2]

264000K402 ..(KA's)

ANSWER: 050 (1.00)

d.

REFERENCE:

OTM SS282AR3, P 20 OF 65, LO #5

KA: 212000K502 [3.3/3.4]

212000K502 ..(KA's)

ANSWER: 051 (1.00)

b.

REFERENCE:

SSM 25D pg. 23. LO 9

K/A: 215005K505 [3.6/3.6]

215005K505 ..(KA's)

ANSWER: 052 (1.00)

a.

REFERENCE:

AOP 19.0, pg 3

K/A: 295018K202 [3.4/3.6]

295018K202 ..(KA's)

ANSWER: 053 (1.00)

c.

REFERENCE:

AOP 20.0, pg 4

K/A: 295019K214 [3.2/3.2]

295019K214 ..(KA's)

ANSWER: 054 (1.00)

d.

REFERENCE:

AOP-36.1, pg 3 and 4

K/A: 295003G003 [3.9/4.1]

295003G003 ..(KA's)

ANSWER: 055 (1.00)

a.

REFERENCE:

AOP-04.0 pg 4

K/A: 202002G014 [3.9/3.5]

202002G014 ..(KA's)

ANSWER: 056 (1.00)

c.

REFERENCE:

AOP-14.0, pg 3 and 4

K/A: 295010A206 [3.6/3.6]

295010A206 ..(KA's)

ANSWER: 057 (1.00)

c.

REFERENCE:

AOP-23.0, pg 3

K/A: 295009G011 [4.3/4.5]

295009G011 ..(KA's)

ANSWER: 058 (1.00)

b.

REFERENCE:

AOP-4.1, pg 3

K/A: 295014G010 [4.0/3.9]

295014G010 ..(KA's)

ANSWER: 059 (1.00)

d.

REFERENCE:

AOP - 16.0 Rev. 7, P. 3 OF 6

KA: 295006G010 [4.1/4.2]

295006G010 ..(KA's)

ANSWER: 060 (1.00)

a.

REFERENCE:

AOP-12.0, REV 4, P. 3 of 4

KA: 295004G011 [3.7/3.9]

295004G011 ..(KA's)

ANSWER: 061 (1.00)

d.

REFERENCE:

OPS-CLS-SM-300-C, REV 0, P 12 OF 35

KA: 295037G007 [3.4/3.5]

295008K208 ..(KA's)

ANSWER: 062 (1.00)

b.

REFERENCE:

AOP 04.4, Rev. 4, P. 3 of 4

KA: 295001A101 [3.5/3.6]

295001A101 ..(KA's)

ANSWER: 063 (1.00)

d.

REFERENCE:

BSEP/Vol. II/SD-18, Rev 12

KA: 295031K203 [4.2/4.3]

295031K203 ..(KA's)

ANSWER: 064 (1.00)

c.

REFERENCE:

EOP 2 & 3

KA: 295024G011 [3.9/3.7]

295024G011 ..(KA's)

ANSWER: 065 (1.00)

d. [+1.0]

REFERENCE:

Malfunction C&E #267, Rev 22, P. 1 of 3
SML-4-C 14-C, pg 33, LO 28e

KA: 295008A105 [3.3/3.3]

295008A105 ..(KA's)

ANSWER: 066 (1.00)

d.

REFERENCE:

Malfunction C&E #239, Rev 24, P. 1 of 2
KA: 295009A201 [4.2/4.2]
295009A201 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

Malfunction C&E #263, Rev 13, P. 1 of 2
SSM 14-2B, pg 22, LO 3j

KA: 206000K603 [2.9/3.1]

206000K603 ..(KA's)

ANSWER: 068 (1.00)

d.

REFERENCE:

Malfunction C&E #297, Rev 13, P. 2 of 3

KA: 295003A101 [3.7/3.8]

295003A101 ..(KA's)

ANSWER: 069 (1.00)

c.

REFERENCE:

BSEP/Vol. II/SD-46, Rev 10, P. 2

KA: 295019K303 [3.2/3.2]

295019K303 ..(KA's)

ANSWER: 070 (1.00)

a.

REFERENCE:

AOP-37.0, pg 4

K/A: 295002G010 [3.8/3.7]

295002G010 ..(KA's)

ANSWER: 071 (1.00)

a.

REFERENCE:

Malfunction C&E #324, Rev 21, P. 1 of 2
AOP-37.0, pg 3

KA: 295002K201 [3.5/3.5]

295002K201 ..(KA's)

ANSWER: 072 (1.00)

b.

REFERENCE:

SDM 15-e, p. 28

KA: 295021A206 [3.2/3.3]

295021A206 ..(KA's)

ANSWER: 073 (1.00)

c.

REFERENCE:

SMM 28-2-A, p. 39-40, LO 5.

KA: 295006A206 [3.5/3.8]

295006A206 ..(KA's)

ANSWER: 074 (1.00)

a.

REFERENCE:

OPS-CLS-SM-300E, p. 50

KA: 295037K303 [4.1/4.5]

295037K303 ..(KA's)

ANSWER: 075 (1.00)

b.

REFERENCE:

SSM 18-2A, p. 26-27, L.O. 7.

K/A: 245000A301 [3.6/3.6]

245000A301 ..(KA's)

ANSWER: 076 (1.00)

d.

REFERENCE:

LPS-CLS-SM-300J, p. 12-14.

KA: 295015K201 [3.8/3.9]

295015K201 ..(KA's)

ANSWER: 077 (1.00)

d.

REFERENCE:

AOP 32,p. 3.

K/A: 295016G010 [3.8/3.6]

295016G010 ..(KA's)

ANSWER: 078 (1.00)

d.

REFERENCE:

AOP 15, p. 6

K/A: 295021K305 [3.6/3.8]

295021K305 ..(KA's)

ANSWER: 079 (1.00)

c.

REFERENCE:

EOP-01, "Reactor Scram Procedure", Step RSP-006

KA: 295005A208 [3.2/3.3]

295005A208 ..(KA's)

ANSWER: 080 (1.00)

c.

REFERENCE:

AOP-15.0, "Alternate Shutdown Cooling Methods", Rev. 5, Page 7

KA: 295021A201 [3.5/3.6]

295021A201 ..(KA's)

ANSWER: 081 (1.00)

d.

REFERENCE:

SSM 29-2A, p. 34, LO 5.

KA: 295023G007 [2.9/3.6]

295023G007 ..(KA's)

ANSWER: 082 (1.00)

d.

REFERENCE:

SML-LOI-CLS-SM-300-L, pg 71, LO 16

K/A: 295028K101 [3.5/3.7]

295028K101 ..(KA's)

ANSWER: 083 (1.00)

a.

REFERENCE:

OPS-CLS-SSM-304-1, pg 18, LO 1c. and d.

K/A: 295016K303 [3.5/3.7]

295016K303 ..(KA's)

ANSWER: 084 (1.00)

c.

REFERENCE:

10CFR20

K/A: 294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 085 (1.00)

a.

REFERENCE:

PEP-02.4, Emergency Control - Site Area Emergency, Rev.14, Section 1.1.10

K/A: 294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

Radiation Control and Protection Manual, Rev. 21, Section 7.1.4

K/A: 294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 087 (1.00)

b.

REFERENCE:

OE&RC-0250, Posting of Areas/Materials, Rev. 19, Section 10.6

K/A: 294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 088 (1.00)

a.

REFERENCE:

OI-13, Valve and Electrical Lineup Controls, Rev. 35, Section 4.0.1

K/A: 294001K101 [3.7/3.7]

294001K101 ..(KA's)

ANSWER: 089 (1.00)

c.

REFERENCE:

OI-13, Valve and Electrical Lineup Controls, Rev. 35, Section 4.2.1.B

K/A: 294001K101 [3.7/3.7]

294001K101 ..(KA's)

ANSWER: 090 (1.00)

c.

REFERENCE:

AI-107, Instructions For Working in Hot Environments, Rev. 5, Section 7.4

K/A: 294001K108 [3.1/3.4]

294001K108 ..(KA's)

ANSWER: 091 (1.00)

d.

REFERENCE:

AI-59, Jumpering and Wire Removal, Rev. 21, Section 5.1.1

K/A: 294001K107 [3.3/3.6]

294001K107 ..(KA's)

ANSWER: 092 (1.00)

b.

REFERENCE:

PEP-03.8.1, Evacuation, Rev. 3, Section 3.2.2

K/A: 294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 093 (1.00)

c.

REFERENCE:

OI-13, "Valve & Electrical Lineup Administrative Controls", Rev. 034, Page 7

K/A: 294001K104 [3.3/3.6]

294001K104 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

OI-05, "Annunciator Status", Rev. 020, pages 1, ?

K/A: 294001A113 [4.5/4.3]

294001A113 ..(KA's)

ANSWER: 095 (1.00)

b.

REFERENCE:

10CFR50.54 (x) and (y)
Licensed Operator Training 07-D, page 29, Obj. 4

K/A: 294001A111 [3.3/4.3]

294001A111 ..(KA's)

ANSWER: 096 (1.00)

d.

REFERENCE:

Student Study Material HO-07-2/3-D3, page 6

K/A: 294001K114 [3.2/3.4]

294001K114 ..(KA's)

ANSWER: 097 (1.00)

c.

REFERENCE:

AOP 5.0, Radioactive Spills, High Radiation and Airborne Activity, Rev. 6, Sections 3.1(3) and 4.0

K/A: 295017G010 [3.9/3.8]

295017G010 ..(KA's)

ANSWER: 098 (1.00)

d.

REFERENCE:

LOI-CLS-SM-300-N, Rev. 0, page 5
EOP-4, Radioactive Release Control

K/A: 295017G011 [4.2/4.5]

295017G011 ..(KA's)

ANSWER: 099 (1.00)

d.

REFERENCE:

LOI-CLS-SM-300-L, page 97, Obj. 3d
EOP-2, Primary Containment Control Procedure

K/A: 295029K301 [3.5/3.9]

295029K301 ..(KA's)

ANSWER: 100 (1.00)

a.

REFERENCE:

SD-12, Primary Containment Isolation system

KA: 295020A101 [3.6/3.6]

295020A101 ..(KA's)

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

ANSWER KEY

MULTIPLE CHOICE

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

ANSWER KEY

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

ANSWER KEY

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

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

R O Exam B W R Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	8000001
002	1.00	8000002
003	1.00	8000003
004	1.00	8000004
005	1.00	8000005
006	1.00	8000006
007	1.00	8000007
008	1.00	8000008
009	1.00	8000010
010	1.00	8000011
011	1.00	8000012
012	1.00	8000014
013	1.00	8000015
014	1.00	8000016
015	1.00	8000017
016	1.00	8000018
017	1.00	8000019
018	1.00	8000020
019	1.00	8000021
020	1.00	8000022
021	1.00	8000023
022	1.00	8000024
023	1.00	8000025
024	1.00	8000026
025	1.00	8000027
026	1.00	8000028
027	1.00	8000029
028	1.00	8000030
029	1.00	8000031
030	1.00	8000032
031	1.00	8000033
032	1.00	8000034
033	1.00	8000035
034	1.00	8000036
035	1.00	8000037
036	1.00	8000038
037	1.00	8000040
038	1.00	8000041
039	1.00	8000042
040	1.00	8000043
041	1.00	8000044
042	1.00	8000045
043	1.00	8000046
044	1.00	8000047
045	1.00	8000048
046	1.00	8000049
047	1.00	8000050
048	1.00	8000051
049	1.00	8000052

RO Exam BWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
050	1.00	8000053
051	1.00	8000054
052	1.00	8000058
053	1.00	8000059
054	1.00	8000060
055	1.00	8000062
056	1.00	8000063
057	1.00	8000064
058	1.00	8000065
059	1.00	8000066
060	1.00	8000067
061	1.00	8000068
062	1.00	8000070
063	1.00	8000072
064	1.00	8000075
065	1.00	8000076
066	1.00	8000077
067	1.00	8000078
068	1.00	8000079
069	1.00	8000080
070	1.00	8000082
071	1.00	8000083
072	1.00	8000085
073	1.00	8000087
074	1.00	8000090
075	1.00	8000092
076	1.00	8000093
077	1.00	8000095
078	1.00	8000096
079	1.00	8000097
080	1.00	8000098
081	1.00	8000099
082	1.00	8000104
083	1.00	8000105
084	1.00	8000107
085	1.00	8000108
086	1.00	8000109
087	1.00	8000110
088	1.00	8000111
089	1.00	8000112
090	1.00	8000114
091	1.00	8000116
092	1.00	8000119
093	1.00	8000120
094	1.00	8000121
095	1.00	8000122
096	1.00	8000123
097	1.00	8000124
098	1.00	8000125

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

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

QUESTION	VALUE	REFERENCE
099	1.00	8000127
100	1.00	8000128

	100.00	

	100.00	

R O Exam BWR Reactor

Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
095	1.00	294001A111
094	1.00	294001A113
092	1.00	294001A116
085	1.00	294001A116
088	1.00	294001K101
089	1.00	294001K101
087	1.00	294001K103
086	1.00	294001K103
084	1.00	294001K103
093	1.00	294001K104
091	1.00	294001K107
090	1.00	294001K108
096	1.00	294001K114

PWG Total	13.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
001	1.00	201001A204
041	1.00	201002A402
042	1.00	201002K404
055	1.00	202002G014
017	1.00	203000A304
018	1.00	203000K117
016	1.00	206000K403
067	1.00	206000K603
020	1.00	209001A209
021	1.00	209001K601
011	1.00	211000A402
044	1.00	212000A215
043	1.00	212000K412
050	1.00	212000K502
024	1.00	215003A401
025	1.00	215003K401
022	1.00	215004K401
023	1.00	215004K602
026	1.00	215005K103
051	1.00	215005K505
015	1.00	217000G007
014	1.00	217000G010
013	1.00	218000K101
009	1.00	223002A102

R O Exam BWR Reactor

Organized by KA Group

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
030	1.00	259001K403
031	1.00	259002A201
008	1.00	261000K601
049	1.00	264000K402

PS-I Total	28.00	

Group II

QUESTION	VALUE	KA
005	1.00	202001K101
006	1.00	202001K117
047	1.00	204000K104
007	1.00	204000K401
048	1.00	204000K404
027	1.00	215002A203
019	1.00	230000K403
003	1.00	239001K127
002	1.00	239001K201
075	1.00	245000A301
029	1.00	256000A202
045	1.00	262001A401
046	1.00	262001K301
033	1.00	271000A301
032	1.00	271000G004
034	1.00	272000G007
035	1.00	272000K603
039	1.00	286000A301
040	1.00	290003A301

PS-II Total	19.00	

Group III

QUESTION	VALUE	KA
028	1.00	215001K401
037	1.00	234000K402
038	1.00	234000K502
036	1.00	268000K106

PS-III Total	4.00	

R O Exam BWR Reactor

Organized by KA Group

PLANT SYSTEMS

QUESTION	VALUE	KA
PS Total	51.00	

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
079	1.00	295005A208
073	1.00	295006A206
059	1.00	295006G010
012	1.00	295007A104
066	1.00	295009A201
057	1.00	295009G011
004	1.00	295009K301
056	1.00	295010A206
058	1.00	295014G010
076	1.00	295015K201
064	1.00	295024G011
063	1.00	295031K203
074	1.00	295037K303

EPE-I Total	13.00	

Group II

QUESTION	VALUE	KA
062	1.00	295001A101
070	1.00	295002G010
071	1.00	295002K201
068	1.00	295003A101
054	1.00	295003G003
060	1.00	295004G011
065	1.00	295008A105
061	1.00	295008K208
077	1.00	295016G010
083	1.00	295016K303
097	1.00	295017G010
098	1.00	295017G011
052	1.00	295018K202
053	1.00	295019K214
069	1.00	295019K303
100	1.00	295020A101
082	1.00	295028K101
099	1.00	295029K301

R O Exam BWR Reactor

Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
010	1.00	295030K103

EPE-II Total	19.00	

Group III

QUESTION	VALUE	KA
080	1.00	295021A201
072	1.00	295021A206
078	1.00	295021K305
081	1.00	295023G007

EPE-III Total	4.00	

EPE Total	36.00	

Test Total	100.00	

Brunswick 94-300
Master

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR OPERATOR LICENSE
REGION 2

CANDIDATE'S NAME: MASTER

FACILITY: Brunswick 1 & 2

REACTOR TYPE: BWR-GE4

DATE ADMINISTERED: 94/01/24

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	______ 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

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

A N S W E R   S H E E T

Multiple Choice (Circle or X your choice)

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

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

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

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

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

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

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

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

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

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

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

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

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

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

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

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

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QUESTION: 001 (1.00)

A reactor scram has occurred. The equalizing valves in the Control Rod Drive Hydraulic System have failed to open. WHICH ONE (1) of the following is the consequence of the failure?

a. Control rod movement could cause excessive control rod speed.

b. Control rods cannot be moved.

c. The scram discharge volume could be pressurized to reactor pressure.

d. The drive water header pressure could exceed 300 psi.

QUESTION: 002 (1.00)

While operating the plant at 30% power the DC power is lost to MSIV 1B21-FO28C. WHICH ONE (1) of the following is the effect on the MSIV?

a. The slow closure capability of the valve will be lost and the valve will maintain position.

b. The slow closure capability of the valve will be lost and the valve will close.

c. The position indication will be lost and the valve will maintain position.

d. The position indication will be lost and the valve will close.

QUESTION: 003 (1.00)

The following conditions exist for unit 1.

The Mode Switch is in RUN
Reactor power 10%
MSIV 1B21-FO22C is stuck at 25% open.

WHICH ONE (1) of the following MSIVs, if closed, will cause a Half Scram signal?

a. 1B21-F022D

b. 1B21-FO28C

c. 1B21-FO22B

d. 1B21-FO28B

QUESTION: 004 (1.00)

When reactor vessel water level is less than 182" AND reactor feed pump flow is less than 20% rated flow for either pump, the recirculation flow control speed limiter #2 (45% demand) is in effect. WHICH ONE (1) of the following is the purpose of the speed limiter #2?

a. To prevent running the recirculation pump at high speed with the discharge valve only partially open.

b. To provide NPSH protection for the recirculation jet pumps at low feedwater flow.

c. To limit reactor power so that the feedwater system will be able to maintain or recover reactor water level on loss of a reactor feed pump.

d. To limit recirculation flow such that possible operation in the areas of instability is reduced.

QUESTION: 005 (1.00)

The Standby Gas Treatment System is in standby. WHICH ONE (1) of the following systems can fail and leave the SBGT in a condition in which it will automatically start and perform its intended function?

a. Process Radiation Monitoring System

b. 480 Volt AC System

c. Reactor Building Ventilation System

d. Interruptible Air Systems

QUESTION: 006 (1.00)

As SRO you receive a call that some of the piping for the non-interruptable instrument air (RNA) had been damaged. The caller did not know what was being supplied by the effected piping. WHICH ONE (1) of the following RB HVAC components could be effected by the loss of RNA?

a. Supply and exhaust fan discharge dampers.

b. The two position dampers in the ventilation supply lines to the refuel floor.

c. The variable position dampers in the exhaust lines from the refuel floor.

d. The reactor building isolation dampers.

QUESTION: 007 (1.00)

A valid Group 8 isolation has occurred. Other plant parameters are as follows:

Reactor steam dome pressure is 125 psig.
Drywell pressure is 2.1 psig.
Reactor building exhaust Rad. is 2 mr/hr.
Reactor building exhaust temperature is 90 deg. F.

WHICH ONE (1) of the following groups of components HAVE ISOLATED?

a. Post Accident Sampling System valves AND HPCI Turbine Exhaust Vacuum Breaker (E41-F075)

b. Containment Atmospheric Control System valves AND RNA-SV-5261 and RNA-SV-5262, Div. 1 and Div. 2 RX Building Non-Interruptible Air to Drywell Isolation Valve

c. HPCI Turbine Exhaust Vacuum Breaker (E41-F075) AND RCIC Exhaust Line Vacuum Breaker (E51-F062)

d. RCIC Exhaust Line Vacuum Breaker (E51-F062) AND RNA-SV-5261 and RNA-SV-5262, Div. 1 RX Building Non-Interruptible Air to Drywell Isolation Valve

QUESTION: 008 (1.00)

Assuming the level in the suppression pool remains above the elevation of the downcomer openings, WHICH ONE (1) of the following is the Heat Capacity Level Limit?

a. The highest suppression pool level after RPV depressurization that will NOT cover the downcomer vacuum breakers.

b. The lowest suppression pool level after RPV depressurization that will NOT cover the downcomer vacuum breakers.

c. The highest suppression pool water level at which initiation of RPV depressurization will NOT result in exceeding the Heat Capacity Temperature Limit.

d. The lowest suppression pool water level at which initiation of RPV depressurization will NOT result in exceeding the Heat Capacity Temperature Limit.

QUESTION: 009 (1.00)

WHICH ONE (1) of the following describes the operation of the Standby Liquid Control (SLC) pumps and/or squib valves?

a. When the SLC pumps are started from the RTGB control switch, depending on switch position, one pump will start and one squib valve will fire.

b. When the SLC pumps are started from the RTGB control switch, depending on switch position, one pump will start and both squib valves will fire.

c. When the SLC system is initiated from the RTGB, the pumps will only start if the local control switches are in the AUTO position and both squib valves will fire.

d. When the SLC pumps are started at the local pushbutton station the squib valves will fire and the RWCU will remain unaffected.

QUESTION: 010 (1.00)

WHICH ONE (1) of the following sets of conditions for the Standby Liquid Control System meets the requirements of Tech Specs for operability? (See Reference)

a. Sodium pentaborate (% by weight) - 15%
Tank volume - 3000 gallons
Solution temperature - 60F

b. Sodium pentaborate (% by weight) - 14.5%
Tank volume - 2500 gallons
Solution temperature - 75F

c. Sodium pentaborate (% by weight) - 16%
Tank volume - 4000 gallons
Solution temperature - 90F

d. Sodium pentaborate (% by weight) - 20%
Tank volume - 3500 gallons
Solution temperature - 80F

QUESTION: 011 (1.00)

During a transient RPV pressure has increased to 1120 psig and started to fall due to the opening of SRVs. RPV level has remained in the normal operating range. Assume no operator action and the SRVs opened at their setpoint.

WHICH ONE (1) of the following describes the number of OPEN SRVs?

a. Four

b. Seven

c. Eight

d. Eleven

QUESTION: 012 (1.00)

All the required conditions have been met for ADS initiation and depressurization is in progress.

WHICH ONE (1) of the following describes how the Automatic Depressurization System (ADS) will be affected if all the RHR and Core Spray Pumps trip off?

a. Depressurization stops; manual initiation using pushbuttons is required to re-establish depressurization when a Core Spray or RHR pump is restored.

b. Depressurization stops; automatic depressurization will be reestablished when a core spray or both pumps in a RHR loop are restored.

c. Depressurization continues without pumps running.

d. Depressurization continues for 105 sec and if no RHR or core spray pumps are back in service, then depressurization stops.

QUESTION: 013 (1.00)

The reactor is operating when a transient occurs that automatically initiates the RCIC system. Following the initiation an isolation signal automatically isolates the RCIC system. No manual actions have been taken on the RCIC system except the operator resets the isolation logic when the isolation signal clears and resets the trip and throttle valve. WHICH ONE (1) of the following is the expected system response providing the initiating signal exists?

a. The system will reisolate on a "RCIC Turbine Exhaust Diaphragm High Pressure" signal.

b. The RCIC turbine will trip on overspeed.

c. Water hammer will occur in the RCIC pump discharge line.

d. The RCIC turbine will remain shut down.

QUESTION: 014 (1.00)

You are performing a test on the RCIC isolation logic when you place the RCIC Logic A selector switch to the TEST position. WHICH ONE (1) of the following signals will be bypassed for that division?

a. RCIC Equipment Room High Differential Temperature

b. RCIC Turbine Exhaust Diaphragm High Pressure

c. RCIC Steam line Low Flow

d. RCIC Low Pump Suction Pressure

QUESTION: 015 (1.00)

WHICH ONE (1) of the following HPCI turbine trips is required to be manually reset by the operator. Assume in each choice that the HPCI turbine was tripped from the stated parameter and all parameters are now in the normal operating range.

a. Reactor High water Level

b. HPCI Pump Low Suction Pressure

c. Turbine High Exhaust Pressure

d. Overspeed

QUESTION: 016 (1.00)

WHICH ONE (1) of the following pressures is the point at which the Residual Heat Removal System operating in the Low Pressure Coolant Injection (LPCI) mode will BEGIN to inject water into the reactor recirculation system as reactor pressure lowers?

a. 425 psig

b. 395 psig

c. 195 psig

d. 115 psig

QUESTION: 017 (1.00)

The following plant conditions exist:

Reactor pressure	900 psig
Reactor level	185 inches

WHICH ONE (1) of the following describes the LPCI injection valves that can be opened with NO LPCI injection signal present?

a. F015A ONLY

b. F017A ONLY

c. BOTH F015A AND F017A at the same time

d. EITHER F015A OR F017A but not at the same time

QUESTION: 018 (1.00)

The plant is in normal operation when a break occurs in the core spray piping inside the reactor vessel between the core shroud and the vessel wall. WHICH ONE (1) of the following describes the behavior of the core spray sparger line break detection?

The core plate/core spray sparger differential pressure indication will:

a. move from negative pressure towards more negative pressure.

b. move from positive pressure towards more positive pressure.

c. move from positive pressure towards negative pressure.

d. move from negative pressure towards positive pressure.

QUESTION: 019 (1.00)

The core spray system has initiated on a valid signal. It is running normally when there is a loss of AC power to pump 1A. This results in the pump tripping. WHICH ONE (1) of the following is the response of the pump when AC power is restored?

a. The pump must be manually restarted.

b. The pump will automatically restart immediately.

c. The pump will automatically restart after a 10 second time delay.

d. The pump will automatically restart after a 15 second time delay.

QUESTION: 020 (1.00)

During a reactor start up you find indication that the SRM system has lost monitoring capability and you receive a rod block. The SRM drive control circuits are still operable. WHICH ONE (1) of the following power sources has been lost?

a. 24 Volt DC

b. 120 Volt AC

c. RPS bus

d. UPS

QUESTION: 021 (1.00)

During reactor power ascension the IRM reading for one channel is:

Numerical value 20
Range 5
Red Scale range (0-40)

You have been directed to range up this channel to Range 6. WHICH ONE (1) of the following is the expected numerical value on Range 6, (0-125)?

a. 20

b. 51

c. 60

d. 62

QUESTION: 022 (1.00)

WHICH ONE (1) of the following is the protective action when the reactor mode switch is in STARTUP and there are fewer than 11 LPRM detectors in OPERATE for APRM channel "C"? The shorting links are installed.

a. Rod select block

b. Half SCRAM on RPS A

c. Half SCRAM on RPS B

d. Full SCRAM

QUESTION: 023 (1.00)

Unit 2 is operating at 100% power with the FWLCS in three element control. WHICH ONE (1) of the following failures would cause a reduction of NPSH to the recirculation pumps?

a. Loss of one feed flow input to FWLCS

b. The selected level instrument fails low

c. Loss of all feed flow inputs to FWLCS

d. One steam flow signal fails low

QUESTION: 024 (1.00)

WHICH ONE (1) of the following predicts the automatic response of the Augmented Off-Gas System? Assume each parameter (one-at-a-time) is above the trip setpoint.

	H2 Downstream of Recombiner (High)	Moisture Down-stream of Guard Bed (High)	System Flow (Hi-Hi)
a.	Bypass and Isolate	Bypass and Isolate	Bypass only
b.	Bypass only	Bypass and Isolate	Bypass only
c.	Bypass and Isolate	Bypass only	Bypass and Isolate
d.	Bypass only	Bypass only	Bypass and Isolate

QUESTION: 025 (1.00)

WHICH ONE (1) of the following power sources supplies the main stack radiation monitors, recorders and sample skids?

a. RPS distribution panel

b. Vital 120 VAC distribution panel 2AB-RX

c. Unit 2, 120V UPS distribution panel 2A

d. Vital 480 VAC panel MCC-2XG

QUESTION: 026 (1.00)

The ~~reactor building~~ Rad Waste floor drain collector tank level is increasing at an abnormally high rate. WHICH ONE (1) of the following could be the cause for the level increasing?

a. RWCU gland seal leak off is excessively high.

b. Excessive water leakage in the HPCI pump area.

c. Excessive recirculation pump seal leakoff.

d. RHR service water leaks.

QUESTION: 027 (1.00)

During refueling the refueling SRO recognized that the frame mounted hoist "LIFT" function deenergized. WHICH ONE (1) of the following can cause this event to occur? Rod 22-31 is at position 02 and the mode switch is in REFUEL.

a. A second control rod is withdrawn.

b. The main grapple is loaded with fuel over the core.

c. The mode switch is repositioned to STARTUP.

d. The frame mounted hoist is loaded with fuel near the core.

QUESTION: 028 (1.00)

During normal operation the fire system pressure dropped to 93 psig and then increased to 126 psig. WHICH ONE (1) of the following describes the fire system pump status if there NO operator action taken?

a. Jockey pumps are running, diesel driven fire pump is running and the motor driven fire pump is running.

b. Jockey pumps are off, diesel driven fire pump is off and the motor driven fire pump is running.

c. Jockey pumps are running, diesel driven fire pump is off and the motor driven fire pump is running.

d. Jockey pumps are off, diesel driven fire pump is running and the motor driven fire pump is off.

QUESTION: 029 (1.00)

WHICH ONE (1) of the following describes the actions that occur when the control room HVAC system detects a high chlorine condition during normal operations?

a. The control room remains in the normal mode except the normal supply and exhaust are isolated.

b. The control room goes into a recirculation mode and only the normal supply path isolates.

c. The control room goes into a recirculation mode bypassing the HEPA filters and the cable spreading room ventilation dampers isolate and the mechanical equipment ventilation damper isolates.

d. The control room remains in the normal mode, supply and exhaust paths isolate along with the mechanical equipment room dampers.

QUESTION: 030 (1.00)

While a control rod is being inserted using the EMERGENCY IN position of the ROD OUT NOTCH OVERRIDE SWITCH, rod motion stops. WHICH ONE (1) of the following could have terminated rod insertion?

a. the automatic sequence timer deenergizes

b. loss of power to the "withdraw" Directional Control Valve

c. a RWM select block

d. a RWM insert block

QUESTION: 031 (1.00)

WHICH ONE (1) of the following describes the MINIMUM parameter(s) for which the Rod Worth Minimizer System is completely and automatically BYPASSED?

a. Steam flow and feed flow greater than 35%.

b. Steam flow greater than 20%.

c. Steam flow greater than 35%.

d. Steam flow and feed flow greater than 20%.

QUESTION: 032 (1.00)

The reactor is shutdown with IRMs A, F, and G on range 3, the plant experiences an IRM HI-HI condition and a reactor scram signal is generated. WHICH ONE (1) of the following identifies possible mode switch position(s) for this event?

a. Refuel only

b. Startup only

c. Startup and Refuel

d. Refuel, Startup and Run

QUESTION: 033 (1.00)

A 4160 VAC breaker has been tested and the clearance removed. The breaker passed the test (charging spring motor toggle to OFF, breaker closed and then opened) and racked in. WHICH ONE (1) of the following is the result of the operator failing to reclose the charging motor toggle switch?

a. The breaker has normal control power indication on the control panel and can be closed and opened one time.

b. The breaker has no control power indication on the control panel and can be closed and opened one time.

c. The breaker has normal control power indication on the control panel but cannot be operated from the control panel.

d. The breaker has no control power indication on the control panel and cannot be operated from the control panel.

QUESTION: 034 (1.00)

WHICH ONE (1) of the following system loads are affected by a loss of power to E-5 480 Volt bus?

a. Diesel Generator #2 Jacket Water Cooling Pump

b. UPS Unit 1 Alternate Feed

c. DG Vent Exhaust Fan B

d. Loop 1A Core Spray Valves

QUESTION: 035 (1.00)

WHICH ONE (1) of the following would be a source of radioactive leakage into the Reactor Building Closed Cooling Water System?

a. Reactor Water Cleanup

b. Control Rod Drive

c. Penetration Cooling

d. Reactor Feedwater

QUESTION: 036 (1.00)

WHICH ONE (1) of the following isolation signals will NOT close RWCU Outboard Isolation Valve (G31-F004)?

a. Low Level #2

b. NRHX High Outlet Temperature

c. High Drywell Pressure

d. SLC Initiation

QUESTION: 037 (1.00)

The Diesel Generators have started in response to a LOCA signal. WHICH ONE (1) of the following signals will shutdown the Diesel Engines?

a. Lube oil pressure - 22 psig

b. Lube oil temperature - 197 deg F

c. Jacket water pressure - 15 psig

d. Jacket water temperature - 187 deg F

QUESTION: 038 (1.00)

With reactor power less than 2% and the mode switch in STARTUP, WHICH ONE (1) of the following inputs to the RPS is bypassed?

a. IRM Hi Hi.

b. Flow biased APRM

c. Main Steam Line Hi Rad.

d. Turbine stop valve closure

QUESTION: 039 (1.00)

The following reactor conditions exist:

Core flow is 37%

Reactor Power is 22%

Recirculation flow is 20%

WHICH ONE (1) of the following is the maximum flow biased APRM SCRAM setpoint required by Technical Specifications?

a. Less than or equal to 55.2%

b. Less than or equal to 77.2%

c. Less than or equal to 78.5%

d. Less than or equal to 90.4%

QUESTION: 040 (1.00)

The reactor is operating at full power when the MSIVs close and cause a SCRAM. Which one of the following is the maximum allowable suppression chamber temperature allowed by Technical Specifications?

a. 95 degrees F.

b. 105 degrees F.

c. 110 degrees F.

d. 120 degrees F.

QUESTION: 041 (1.00)

While operating the plant at full power, the discharge bypass valve for recirculation loop "A" is declared INOPERABLE. The valve is now in the CLOSED position. WHICH ONE (1) of the following is the required action?

a. Demonstrate the bypass valve is OPERABLE within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. Demonstrate the bypass valve is OPERABLE within 12 hours or be in at least HOT SHUTDOWN within the next 24 hours.

c. Secure the loop "A" recirculation pump and comply with the applicable Technical Specifications.

d. Verify the valve is closed at least once per 31 days.

QUESTION: 042 (1.00)

The reactor has been operating over an extended period of time and the average primary containment temperature has increased to 140 degrees F. WHICH ONE (1) of the following is the required Technical Specification action?

a. Reduce the average containment air temperature to less than 135 degrees within 8 hours.

b. Initiate hourly monitoring.

c. Be in at least HOT SHUTDOWN within the next 8 hours.

d. Be in at least COLD SHUTDOWN within the next 8 hours.

QUESTION: 043 (1.00)

The reactor is operating with the "A" and "B" CSW pump in service when the CSW header pressure drops to 35 psig. The operator verifies the SWV-3 and SWV-4 are at the throttled position. The "C" CSW pump is INOPERABLE. WHICH ONE (1) of the following actions should be taken?

a. Trip the "A" and "B" CSW pumps and manually SCRAM the reactor.

b. Reduce all Service Water flows to maintain CSW header pressure above 30 psig.

c. Verify the RBCCW containment isolation valves have isolated and manually SCRAM if drywell pressure increases to 2.0 psig.

d. Monitor the CSW header pressure and manually SCRAM the reactor if pressure falls to 30 psig.

QUESTION: 044 (1.00)

The reactor is operating and the Reactor Building Standby Compressor is in AUTO. As air pressure decreases WHICH ONE (1) of the following will start the Reactor Building Standby Compressor?

a. Service air header pressure less than 111 psig.

b. SCRAM pilot valve air header pressure less than 55 psig.

c. Noninterruptible instrument air header pressure less than 95 psig.

d. Pneumatic Nitrogen system supply pressure less than 95 psig.

QUESTION: 045 (1.00)

Both Unit 1 and Unit 2 are operating at full power when the 120 VAC Bus 2E7 is lost. Trouble shooting has been underway for eight hours and it has been determined the bus cannot be restored. WHICH ONE (1) of the following is the correct action?

a. Be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours, Unit 1 only.

b. Be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours, Unit 2 only.

c. Be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours, Unit 1 and Unit 2.

d. Be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours, Unit 1. Manually SCRAM Unit 2.

QUESTION: 046 (1.00)

While operating the reactor at 90% power the "A" recirculation pump experiences a SPEED CONTROL SIGNAL FAILURE. You have ensured this is not a recirculation pump runback and locked the scoop tube on the "A" recirculation pump. You have verified operation in Region "C" with no thermal hydraulic instabilities. WHICH ONE (1) OF the following is the required immediate operator action?

a. INCREASE core flow to greater than 35 Mlbs/hr using the unaffected recirculation pump.

b. DECREASE core flow to less than 35 mlbs/hr using the unaffected recirculation pump.

c. WITHDRAW control rods to the 100% rod line per the Nuclear Engineer's instructions.

d. Continue to monitor for thermal hydraulic instabilities and obtain approval of Nuclear Engineer to insert control rods.

QUESTION: 047 (1.00)

WHICH ONE (1) of the following conditions will NOT be mitigated by the use of procedure EOP-02, "Primary Containment Control"?

a. Primary containment temperature of 140 deg. F.

b. Suppression pool level of -32 inches.

c. Primary containment pressure of 1.5 psig.

d. Primary containment Hv⁻⁻ogen level of 1.8%.

QUESTION: 048 (1.00)

WHICH ONE (1) of the following conditic s a symptom of Condensate and Feedwater failure per AOP-23.0?

a. Safety relief valves on the LP heaters shell side lifting.

b. Indicated reactor vessel level at +190 inches.

c. Steam flow/feed flow mismatch as indicated on the recorder on Panel P603.

d. High offgas flow as indicated the RTGB instrumentation.

QUESTION: 049 (1.00)

During reactor operation you notice reactor power and generator power increasing. One recirculation pump current indication is increasing. Reactor power is 80%, total core flow is 65 million lbs/hr and the recirculation pump mismatch is 15%. WHICH ONE (1) of the following is the immediate operator action?

a. Trip the affected recirculation pump.

b. Lock the scoop tube on the affected recirculation pump.

c. Place the affected recirculation pump in MANUAL control.

d. Manually SCRAM the reactor.

QUESTION: 050 (1.00)

WHICH ONE (1) of the following conditions requires a manual reactor SCRAM if a failure of the RBCCW system occurred?

a. RWCU isolation.

b. High temperature alarm on the RBCCW heat exchanger outlet.

c. Drywell chiller trip.

d. RBCCW pressure less than 60 psig.

QUESTION: 051 (1.00)

WHICH ONE (1) of the following conditions requires entry into an EOP?

a. Failing to reenergize the UPS bus.

b. Primary containment temperature at 131 degrees F.

c. Secondary Containment dp is -0.19 inches H2O.

d. Reactor Building ventilation exhaust radiation level at 2 Mr/hr.

QUESTION: 052 (1.00)

During an ATWS, EOPs direct the operator to inhibit ADS automatic blowdown. WHICH ONE (1) of the following states the basis for this requirement?

a. ADS actuation would in an unacceptable differential pressure across the steam separator.

b. ADS would result in the removal of boron after it has been injected.

c. Core damage could result from a large power excursion if low pressure ECCS systems were to inject.

d. ADS/SRV system flow rate is incapable of assuring fuel cooling through steaming above 3% reactor power.

QUESTION: 053 (1.00)

WHICH ONE (1) of the following is an indication of a jet pump failure that may require entry into AOP 04.4, "Jet Pump Failure"?

a. Decrease in indicated total core flow.

b. Core plate differential pressure decrease.

c. Recirculation loop flow decrease in the loop with the failed jet pump.

d. Recirculation pump discharge pressure increase on the loop with the failed jet pump.

QUESTION: 054 (1.00)

During the execution of EOP-02 "Primary Containment Control", you reach the step to make a decision if the drywell initiation limit is in the safe region. WHICH ONE (1) of the following is the safety significance of the drywell spray initiation limit?

a. Due to convective cooling the pressure reduction is too fast for the primary containment relief system to ensure primary containment integrity.

b. Convective cooling does not have enough pressure suppression capability to suppress primary containment pressure during design basis LOCA events.

c. Due to evaporative cooling the pressure reduction is too fast for the primary containment relief system to ensure primary containment integrity.

d. Evaporative cooling does not have enough pressure suppression capability to suppress primary containment pressure during design basis LOCA events.

QUESTION: 055 (1.00)

The plant has been operating for several days at full power. Reactor coolant system unidentified leakage has been calculated to be 0.5 gpm. The DRYWELL EQUIPMENT DRAIN SUMP LEAK HIGH alarm has just energized. A new unidentified leakage has been calculated and found to be 2.5 gpm. WHICH ONE (1) of the following actions should you take?

a. SCRAM the reactor and conduct an emergency cooldown of the reactor plant.

b. Monitor drywell pressure, temperature and activity and continue operation without restriction.

c. Reduce the leakage to within limits within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

d. Reduce thermal power to less than 25% of RATED THERMAL POWER within the next 4 hours.

QUESTION: 056 (1.00)

Given the following indications:

"NUCLEAR HDR SERV WTR PRESS" annunciates.
"NUCLEAR HDR SW PUMP TRIP" annunciates.
The standby Nuclear Service water pump starts.
Nuclear Service Water HDR pressure is 40 psig.

Which ONE (1) of the following identifies the immediate operator action(s)?

a. Monitor recirculation pump seal temperature, trip recirculation pumps if seals exceed 200F.

b. Isolate non-essential RBCCW heat loads.

c. Start additional Service Water Pumps.

d. Scram the reactor.

QUESTION: 057 (1.00)

Given the following conditions:

Reactor water level +172 inches
Drywell pressure 1.72 psig
Primary containment average temp. 138 F
Reactor Bldg Vent Exhaust Rad Level 3.3 mR/hr
The "Reactor Bldg Vent Rad Hi Exhaust" annunciator has NOT alarmed

WHICH ONE (1) of the following lists of EOPs are you required to enter?

a. EOP-01-RVCP Reactor Vessel Control and EOP-02-PCCP Primary Containment Control.

b. EOP-01-RVCP Reactor Vessel Control and EOP-03-SCCP Secondary Containment Control.

c. EOP-02-PCCP Primary Containment Control and EOP-03-SCCP Secondary Containment Control.

d. EOP-03-SCCP Secondary Containment Control and EOP-04-RRCP Radioactivity Release Control.

QUESTION: 058 (1.00)

During a plant transient, a failure of the RCIC logic bus 'B', due to a ground fault occurs. The RCIC system is manually started. WHICH ONE (1) of the following describes how the RCIC system would respond to RPV water level?

a. It would shut down on high RPV level and would automatically start on a decreasing RPV level.

b. It would NOT shut down on high RPV level and would automatically start on a decreasing RPV level.

c. It would shut down on high RPV level and would NOT automatically start on a decreasing RPV level.

d. It would NOT shut down on high RPV level and would NOT automatically start on a decreasing RPV level.

QUESTION: 059 (1.00)

While operating Unit 2 in a steady state at 78% power the below conditions were observed by the operator:

Total feedwater flow decreased slightly and returned normal.

2B RFP suction flow increased slightly and discharge flow decreased.

Reactor water level decreased slightly and returned to normal.

Reactor power remained relatively constant.

Feedwater level control is in three-element control.

Master level controller is in AUTO

2B RFP controller in AUTO and 2A RFP controller is in MANUAL

WHICH ONE (1) of the following explains the plant response?

a. The selected RPV level transmitter failed and an automatic transfer to the redundant transmitter has occurred.

b. The output signal from 2A RFP controller had decreased slightly an then returned to normal.

c. The control rod pattern has been modified using rods near the outer edges of the core.

d. The 2B RFP minimum flow valve had sensed a low feedwater flow signal and gradually opened and then stayed opened.

QUESTION: 060 (1.00)

During an event which initiated HPCI, WHICH ONE (1) of the following describes how the HPCI system would respond if annunciator A-01 2-5, "HPCI INVRTR POWER FAILURE", went into an alarm condition?

a. HPCI Pump discharge pressure would increase as the control valves open.

b. HPCI steam line drain pot drain valve and condensate discharge valve close.

c. HPCI system logic will sense a low CST level and open the torus suction valves.

d. HPCI discharge flow will decrease and RTGB HPCI flow indication will fail downscale.

QUESTION: 061 (1.00)

Concerning the Service Air system, WHICH ONE (1) of the following describes the reason for the service air isolation valves closing at 105 psig?

a. Assure an adequate supply of breathing air throughout the plant.

b. Assure the minimum allowable pressure is maintained on the Service Air system.

c. Assure an adequate supply of air to the RNA System.

d. Assure an air supply to drywell loads under all expected operating conditions.

QUESTION: 062 (1.00)

A loss of all high pressure injection systems has resulted in RPV level decreasing to TAF. An emergency RPV depressurization has been directed. WHICH ONE (1) of the following states the reason that a minimum of 4 SRVs must be opened?

a. Ensures that the reactor will be depressurized to below ECCS shut off head before the RPV level reaches two thirds core height.

b. Ensures that at the worst case in core life, the APLHGR thermal limit will not be exceeded and inhibit adequate radiant heat transfer.

c. Prevents exceeding 1% plastic strain on the hottest fuel pin in the core allowing fuel cladding failure to release radioactive fission products.

d. Ensures that sufficient steam flow will exist to remove decay heat at low enough pressure for the lowest head ECCS pump to make up for steam flow.

QUESTION: 063 (1.00)

The reactor is operating at full power. As the Reactor Operator you notice the main condenser vacuum is 26 inches Hg and slowly decreasing. The circulating water pumps are verified to be operating properly. WHICH ONE (1) of the following actions is required by AOP-37.0, "Low Condenser Vacuum"?

a. Reduce generator load, as necessary, to maintain condenser vacuum greater than 25 inches Hg.

b. Increase gland sealing steam pressure to greater than 5 psig.

c. Manually SCRAM the reactor.

d. Start BOTH mechanical vacuum pumps.

QUESTION: 064 (1.00)

The reactor is operating at 25% power when the main condenser vacuum decreases to 22" Hg. WHICH ONE (1) of the following events is the direct action? Assume the Mode Switch is in RUN.

a. Main turbine trips.

b. Turbine Bypass Valves close.

c. Group 1 isolation.

d. Automatic reactor SCRAM.

QUESTION: 065 (1.00)

The events listed below have occurred:

Buses 2B, 2C and 2D were being fed from the UAT when the UAT experienced a fault pressure.

Reactor Recirc Pump 2A and 2B MG sets trip.

Condensate Booster Pump 2B experiences a loss of power.

WHICH ONE (1) of the following explains the cause of the plant transient?

a. SAT to 2C breaker failed.

b. SAT to 2D breaker failed.

c. SAT to 2B and 2C failed.

d. SAT to 2B and 2D failed.

QUESTION: 066 (1.00)

Unit 1, "A" loop of RHR, is in shutdown cooling. Plant conditions cause the SDC suction valves, E11-F008 and F009, and the RHR injection isolation valve E11-F015A, to close. Which ONE (1) of the following conditions caused the shutdown cooling isolation?

a. Reactor water level +210 inches

b. Reactor water level +162.5 inches

c. Drywell pressure 2.0 psig

d. Reactor steam dome pressure 140 psig

QUESTION: 067 (1.00)

AOP 32.0, "PLANT SHUTDOWN FROM OUTSIDE CONTROL ROOM" states that when steam flow is less than 3 E6 lbs/hr place the mode switch to shutdown. Which ONE (1) of the following is the reason for waiting until main steam line flow is less than 3 E6 lbs/hr?

a. Prevents high differential pressure across the MSIVs.

b. Prevents pressure transient on vessel internals.

c. Prevents Group 1 isolation during pressure reduction to 700 psig.

d. Prevents MSIV's from immediately shutting on high flow.

QUESTION: 068 (1.00)

Unit 1 has entered AOP-30.0, Safety Relief Valve Failures". Suppression pool temperature has increased to 96 degrees F. WHICH ONE (1) of the following actions should be taken?

a. Enter EOP-02-PCCP, "Primary Containment Control Procedure" only.

b. Cycle the SRV control switch to OPEN and CLOSE or AUTO several times only

c. Cycle the SRV control switch to OPEN and CLOSE or AUTO several times and enter EOP-02-PCCP, "Primary Containment Control Procedure".

d. Runback the recirculation pumps then manually SCRAM the reactor.

QUESTION: 069 (1.00)

In accordance with Technical Specifications, which ONE (1) of the following exceeds a Limiting Safety System Setting?

a. Reactor vessel level drops to +5 inches following a scram, then RCIC restores level to +56.5 inches.

b. With the unit at 20% power, the Main and RFP turbines trip at +208 inches. Reactor scrams on low RPV level.

c. With the unit at 20% power, the EHC pressure regulator fails causing a reactor scram due to MSIV closure at 825 psig.

d. With the unit at 100% power a Group 1 isolation is received on MSL High Radiation. The reactor scrams on MSIV not full open signal.

QUESTION: 070 (1.00)

The lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800F is the definition of:

a. Minimum Zero injection RPV water level

b. Minimum Steam Cooling RPV Water level

c. Maximum Core Uncovery level

d. Minimum RPV Flood level

QUESTION: 071 (1.00)

WHICH ONE (1) of the following sets of plant parameters will directly cause the Main Turbine to trip?

a. Reactor Water Level 191 inches
Turbine Speed 1902 rpm
Condenser Vacuum 25 in Hg
Bearing Oil Supply Header 15 psig

b. Reactor Water Level 210 inches
Turbine Speed 1340 rpm
Condenser Vacuum 24 in Hg
Bearing Oil Supply Header 11 psig

c. Reactor Water Level 161 inches
Turbine Speed 1875 rpm
Condenser Vacuum 26 in Hg
Bearing Oil Supply Header 13 psig

d. Reactor Water Level 155 inches
Turbine Speed 1545 rpm
Condenser Vacuum 23 in Hg
Bearing Oil Supply Header 9 psig

QUESTION: 072 (1.00)

Which ONE (1) of the following methods for alternate rod insertion requires the scram to be reset?

a. Venting the scram air header.

b. Inserting control rods from the Reactor Manual Control System.

c. Venting the over-piston area of the control rod.

d. Using the individual scram test switches.

QUESTION: 073 (1.00)

As a result of a plant accident, the following plant conditions exist:

- The reactor is shutdown due to an earthquake measured at .06 g.
- All rods are fully inserted.
- Unidentified reactor coolant system leakage has increased to 30 gpm.
- RCS activity reported increasing, latest sample is 28 uCi/ml.
- Instantaneous release rate per the OG-06.1 calculation is 1100% of the Technical Specification limit.
- The earthquake resulted in a failure of RHR loop A and the condensate system. Both systems have been successfully isolated.
- One 4 kV E bus lost, remaining buses energized.
- A loss of the NRC red phone and CP&L network is reported.
- One individual injured (not contaminated), medical assistance required to transport to an off-site hospital.

Based on the existing plant conditions, WHICH ONE (1) of the following is the Emergency Action Level classification that would be required? (Appropriate references are available.)

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 074 (1.00)

The Shift Supervisor has determined that a control room evacuation will be required. WHICH ONE (1) of the following is NOT an immediate operator action?

a. Trip the main turbine.

b. Trip both reactor recirculation pumps.

c. Reduce reactor pressure to 700 psig.

d. Place all MSIV switches to AUTO.

QUESTION: 075 (1.00)

WHICH ONE (1) of the following indications is used to monitor plant heatup/cooldown rate while in Alternate Shutdown Cooling?

a. RBCCW heat exchanger outlet temperature.

b. RHR heat exchanger outlet temperature.

c. Safety relief valve tailpipe temperature.

d. The running ECCS pump local suction temperature.

QUESTION: 076 (1.00)

During an emergency evolution, drywell pressure is approximately six psig. This has resulted in the automatic shutdown of the reactor building HVAC system. WHICH ONE (1) of the following conditions must be met in order to assure the safe restarting of the reactor building HVAC system?

a. Drywell pressure below 2.0 psig.

b. The reactor building ventilation exhaust radiation level below 4 mr/hr.

c. Reactor Building pressure is positive.

d. No more than one area temperature is above the Maximum Normal Operating (MNO) temperature.

QUESTION: 077 (1.00)

The temperature control leg of EOP-03, "Secondary Containment Control Procedure", asks whether "a primary system is discharging into the Reactor Building".

WHICH ONE (1) of the following is a Primary System as referenced in this step?

a. Any plant safety-related system required to be operable in Modes 1, 2 and/or 3.

b. Any system whose leak rate will decrease as reactor pressure decreases.

c. Any system required to shutdown the reactor or provide long-term core cooling.

d. Any plant system which penetrates the primary containment.

QUESTION: 078 (1.00)

Concerning the residual heat removal system (RHR):

During reactor operation protective interlocks assure shutdown cooling isolation valves (E11-F008 and F009) remain closed.

WHICH ONE (1) of the following states the reason for the protective interlock?

a. Protects RHR low pressure piping from full reactor pressure.

b. Protects vessel components from severe cool down rates.

c. Protects the RHR pumps from cavitation.

d. Protects against the possibility of draining the RPV.

QUESTION: 079 (1.00)

WHICH ONE (1) of the following defines the Heat Capacity Temperature Limit?

a. The highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the suppression chamber design temperature.

b. The highest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the suppression chamber design pressure.

c. The lowest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the suppression chamber design temperature.

d. The lowest suppression pool temperature at which initiation of RPV depressurization will not result in exceeding the suppression chamber design pressure.

QUESTION: 080 (1.00)

During accident conditions you would like to use the narrow range level instruments to determine RPV level. WHICH ONE (1) of the following sets of parameters are required to determine if the instruments may be used?

a. Drywell temperature and reactor pressure only

b. Indicated level and drywell temperature only

c. Reactor pressure and drywell temperature only

d. Reactor pressure, drywell temperature and indicated level

QUESTION: 081 (1.00)

A fire in the control room of Unit 1 has caused an evacuation of the control room. WHICH ONE (1) of the following actions prevents spurious operation of the SRVs?

a. The supply breakers to 125V DC distribution panels 3A and 3B located on 125/250 switchboards 1A and 1B are opened.

b. DC switchboards 2B/1B loads are stripped except for distribution panels 2B/1B and MCCs 2XDB/1XDB.

c. Battery chargers 1B are transferred to their alternate power supply.

d. De-energize, via keylock switches, the Div. I Nitrogen backup system valves.

QUESTION: 082 (1.00)

A Site Area Emergency has been declared. All emergency response positions are manned and emergency facilities are activated. WHICH ONE (1) of the following duties of the Site Emergency Coordinator can be delegated?

a. Downgrade the emergency classification

b. Terminating the emergency

c. Issuing off-site Protective Action Recommendations

d. Reclassifying the emergency

QUESTION: 083 (1.00)

During a Site Area Emergency, the AO on building rounds did not report to the Control Room and does not respond to the plant paging system. WHICH ONE (1) of the following is the maximum exposure allowed to an individual in order to search for the unaccounted for operator?

a. Brunswick administrative limits

b. 10CFR20 non-emergency limits

c. 25 REM

d. 75 REM

QUESTION: 084 (1.00)

The Emergency Plan has been activated and a Site Area Emergency declared. All Brunswick management personnel are on site and have assumed their positions. WHICH ONE (1) of the following identifies the two individuals who have the authority to approve entry into a radiation field greater than 100 R/hr?

a. Plant General Manager and Environmental and Radiation Control Manager

b. Plant General Manager and Emergency Response Manager

c. Emergency Response Manager and Site Emergency Coordinator

d. Site Emergency Coordinator and Environmental and Radiation Control Manager

QUESTION: 085 (1.00)

Due to the inoperability of a hand/foot monitor, personnel contamination monitoring is being performed with a pancake GM detector. WHICH ONE (1) of the following states the MINIMUM indications at which individuals shall be considered contaminated?

a. any count rate greater than background

b. count rate of 100 cpm above background

c. count rate of 200 cpm above background

d. count rate of 1000 cpm above background

QUESTION: 086 (1.00)

Entry into WHICH ONE (1) of the following areas requires continuous HP monitoring?

a. Locked High Radiation area

b. Restricted High Radiation area

c. High Contamination area

d. Radioactive Materials area

QUESTION: 087 (1.00)

WHICH ONE (1) of the following is indicated by an orange colored handwheel?

a. the handwheel must be turned clockwise to open

b. operation of the valve must be approved by the Shift Supervisor

c. independent verification is required if the valve is operated

d. the valve is a radiological "hot spot"

QUESTION: 088 (1.00)

A level instrument is being returned to service and both the high and low pressure root valves are CLOSED. WHICH ONE (1) of the following states the correct action to perform first to ensure disturbances are not created while returning the instrument to service?

a. CLOSE/check CLOSED the equalizing valve, then open the LOW pressure root valve.

b. CLOSE/check CLOSED the equalizing valve, then open the HIGH pressure root valve.

c. OPEN/check OPENED the equalizing valve, then open the LOW pressure root valve.

d. OPEN/check OPENED the equalizing valve, then open the HIGH pressure root valve.

QUESTION: 089 (1.00)

Unit 1 is in Cold Shutdown with Shutdown Cooling in operation. Unit 2 is in Hot Shutdown and reactor pressure is 105 psig and slowly decreasing. WHICH ONE (1) of the following identifies systems that require preventative maintenance be performed on a 24 hour basis to minimize the length of time they are out of service?

a. Unit 1 HPCI and Unit 2 Core Spray

b. Unit 1 RHR and Unit 1 Core Spray

c. Unit 2 HPCI and Diesel Generator #3

d. Unit 2 RHR and Diesel Generator #1

QUESTION: 090 (1.00)

An operator is assigned to work in a room at 90 degrees F WBGT wearing a single cotton blend plus impervious garment. The task will require and occasional ladder climbing and valving to lineup a system and will take approximately two hours to complete. WHICH ONE (1) of the following is the MAXIMUM time the operator may remain in the environment in order to initiate the task and ensure it is brought to a satisfactory and safe stopping point? (See Attached Procedure AI-107)

a. 25 minutes

b. 30 minutes

c. 37.5 minutes

d. 100 minutes

QUESTION: 091 (1.00)

WHICH ONE (1) of the following actions CANNOT be authorized by approval of a Troubleshooting Control Form (TCF)?

a. equipment realignment outside of a clearance

b. temporary change of a Periodic Test

c. alteration of instrument settings

d. a change in system configuration

QUESTION: 092 (1.00)

Leads for two phases of a three phase motor (Technical Specification-related equipment) have been momentarily lifted. WHICH ONE (1) of the following states the requirements dictated by this action per procedure AI-59, Jumper and Wire Removal?

a. Jumper wire removal tags must be attached

b. Temporary Modification form must be submitted within 24 hours.

c. Independent verification must be performed per drawings

d. Correct motor rotation must be verified upon retermination

QUESTION: 093 (1.00)

All temporary changes to procedures listed in Technical Specifications must have the final approval of WHICH ONE (1) of the following individuals?

a. General Manager

b. Operations Manager

c. NRC Resident Inspector

d. Vice President, Nuclear

QUESTION: 094 (1.00)

A Site Area Emergency has been declared. WHICH ONE (1) of the following is responsible for terminating the event when conditions clear?

a. Site Emergency Coordinator

b. Technical Support Center Manager

c. Emergency Response Manager

d. Emergency Response Manager, State, Counties and NRC jointly

QUESTION: 095 (1.00)

Personnel in the Emergency Organization who do not have a designated emergency assignment when a site evacuation is performed will report to WHICH ONE (1) of the following for instructions/assignments?

a. Control Room Supervisor

b. Operational Support Center Leader

c. Site Emergency Coordinator

d. Immediate Supervisor

QUESTION: 096 (1.00)

WHICH ONE (1) of the following radiation exposure guidelines is the MINIMUM exposure at which the Shift Supervisor may waive the independent verification requirement for a system valve lineup?

a. General area radiation levels are in excess of 10 mr/hr.

b. General area radiation levels are in excess of 100 mr/hr.

c. Total exposure expected during the independent verification is in excess of 10 mr.

d. Total exposure expected during the independent verification is in excess of 100 mr.

QUESTION: 097 (1.00)

WHICH ONE (1) of the following annunciator window colors designates a setpoint important to reactor safety?

a. Red annunciator window with a red bar.

b. Amber annunciator window with a red bar.

c. Red annunciator window with a blue bar.

d. Amber annunciator window with a blue bar.

QUESTION: 098 (1.00)

10CFR50.54 (x) states that reasonable action that departs from a license condition or technical specification may be taken in an emergency to protect the health and safety of the public. WHICH ONE (1) of the following states the MINIMUM approval required?

a. any licensed reactor operator

b. any licensed senior reactor operator

c. Site Emergency Coordinator

d. General Plant Manager

QUESTION: 099 (1.00)

Unit 2 is in Start up at 20% power. WHICH ONE (1) of the following constitutes a loss of Secondary Containment?

a. The Reactor Building normal HVAC is inoperable and isolated.

b. Both Reactor Building ventilation radiation monitors are INOP.

c. A suppression pool/Reactor Building vacuum breaker is open.

d. The Standby Gas Treatment system is inoperable.

QUESTION: 100 (1.00)

A PCIS Group 1 isolation has resulted in a reactor scram on Unit 2. The plant has been stabilized using CRD for level control and SRVs for pressure control. Current plant conditions are as follows:

RPV level 180 inches
RPV pressure 1000 psig
Reactor power IRM range 3
"A" Circulating Water Pump running
"A" Condensate Pump running
"A" Condensate Booster Pump running

WHICH ONE (1) of the following conditions would prevent the reopening of MSIVs?

a. Complete loss of 120 VAC.

b. Condenser vacuum is 0 inches.

c. A malfunction prevents resetting the turbine.

d. SJAE "A" first stage steam supply valve will not open.

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

ANSWER: 001 (1.00)

a.

REFERENCE:

SSM 09-B, pg 8, LO 14.b

K/A: 201001A204 [3.8/3.9]

201001A204 ..(KA's)

ANSWER: 002 (1.00)

c.

REFERENCE:

SSM-17-A, pg. 34, LO10.a.

K/A: 239001K201 [2.8/3.2]

239001K201 ..(KA's)

ANSWER: 003 (1.00)

a.

REFERENCE:

SD-25, "Main Steam System", Rev. 12, Page 5

OSM 17-2A, "Main Steam", Rev. 3, Pages 12 - 14, L.O. - 3

KA: 239001K127[4.0/4.1]

239001K127 ..(KA's)

ANSWER: 004 (1.00)

c.

REFERENCE:

SSM 10-2A, pg 22, LO 3.h.

K/A: 295009K301 [3.2/3.3]

295009K301 ..(KA's)

ANSWER: 005 (1.00)

d.

REFERENCE:

LOI-CLS-LP-010-A, pg 28, LO 9.f.&g.&h.&j.

K/A: 261000K601 [2.9/3.1]

261000K601 ..(KA's)

ANSWER: 006 (1.00)

d.

REFERENCE:

SSM 35-A, pg 26, LO 8.b.

K/A: 288000K603 [2.7/2.7]

288000K603 ..(KA's)

ANSWER: 007 (1.00)

b.

REFERENCE:

SSM 15-e, pg. 64,27,31,30, LO 9-13

K/A: 223002A102 [3.7/3.7]

223002A102 ..(KA's)

ANSWER: 008 (1.00)

d.

REFERENCE:

LOI-CLS-SM-004-A, pg 30, LO 13.b

K/A: 295030K103 [3.8/4.1]

295030K103 ..(KA's)

ANSWER: 009 (1.00)

b.

REFERENCE:

SML-4 14G, pg 10, LO 11

K/A: 211000A402 [4.2/4.2]

211000A402 ..(KA's)

ANSWER: 010 (1.00)

c.

REFERENCE:

T/S 3.1.5
KA: 211000G005 [3.6/4.4]

211000G005 ..(KA's)

ANSWER: 011 (1.00)

c.

REFERENCE:

SSM 14-F, pg. 7, LO 3, 5, 6, and 9

K/A: 295007A104 [3.9/4.1]

295007A104 ..(KA's)

ANSWER: 012 (1.00)

b.

REFERENCE:

SSM 14-F, pg. 14, LO 11

K/A: 218000K101 [4.0/4.0]

218000K101 ..(KA's)

ANSWER: 013 (1.00)

d.

REFERENCE:

SML-4 14-C, pg 44, LO 23

K/A: 217000G010 [3.4/3.5]

217000G010 ..(KA's)

ANSWER: 014 (1.00)

a.

REFERENCE:

SML-4 14-C, pg. 43, LO 18

K/A: 217000G007 [3.8/3.7]

217000G007 ..(KA's)

ANSWER: 015 (1.00)

a.

REFERENCE:

SSM 14-2B, pg. 28, LO 4.1 and 22

K/A: 206000K403 [4.2/4.1]

206000K403 ..(KA's)

ANSWER: 016 (1.00)

c.

REFERENCE:

SSM 14D, pg. 20, LO 20

KA: 203000A304 [3.8/3.7]

203000A304 ..(KA's)

ANSWER: 017 (1.00)

d.

REFERENCE:

SSM 14-D, pg 15, LO 9

K/A: 203000K117 [4.0/4.0]

203000K117 ..(KA's)

ANSWER: 018 (1.00)

d.

REFERENCE:

SSM 14-E, pg 5, LO 10

K/A: 209001A209 [2.9/3.0]

209001A209 ..(KA's)

ANSWER: 019 (1.00)

d.

REFERENCE:

SSM 14-E, pg. 12, LO 15

K/A: 209001K601 [3.4/3.4]

209001K601 ..(KA's)

ANSWER: 020 (1.00)

a.

REFERENCE:

SSM 25-A, pg 16, LO 4

K/A: 215004K602 [3.1/3.3]

215004K602 ..(KA's)

ANSWER: 021 (1.00)

a.

REFERENCE:

SSM 25-B, pg. 6, LO 5

K/A: 215003A401 [3.3/3.3]

215003A401 ..(KA's)

ANSWER: 022 (1.00)

b.

REFERENCE:

SSM 25-D, pg 16, LO 3

K/A: 215005K103 [3.4/3.5]

215005K103 ..(KA's)

ANSWER: 023 (1.00)

d.

REFERENCE:

SML SS172, pg. 26, LO 9

K/A: 259002A201 [3.3/3.4]

259002A201 ..(KA's)

ANSWER: 024 (1.00)

a.

REFERENCE:

SSM 24-2, pg. 13, LO 5.a., c., e.

K/A: 271000A301 [3.3/3.3]

271000A301 ..(KA's)

ANSWER: 025 (1.00)

c. [+1.0]

REFERENCE:

SSM 03-B, Rev. 1, P 10, LO #3 , SD-11 Table 3.5

KA: 272000K603 [2.8/3.0]

272000K603 ..(KA's)

ANSWER: 026 (1.00)

b.

REFERENCE:

SSM 16-A, Rev. 0, P 8, LO #03d

KA:268000K104 [2.7/2.9]

268000K104 ..(KA's)

ANSWER: 027 (1.00)

d.

REFERENCE:

SSP29-2AR4, Rev. 4, P. 41 OF 119, LO #6

KA: 234000K502 [3.1/3.7]

234000K502 ..(KA's)

ANSWER: 028 (1.00)

c.

REFERENCE:

SSM 31-A, Rev. 0, P 27, 28 and 29, LO #9

KA: 286000A301 [3.4/3.4]

286000A301 ..(KA's)

ANSWER: 029 (1.00)

c.

REFERENCE:

SSM 35-B, Augmented, P 3, LO #3

KA: 29003A301 [3.3/3.5]

290003A301 ..(KA's)

ANSWER: 030 (1.00)

d.

REFERENCE:

SML 272A, REV AUG, P 19 and 20, LO #9

KA: 201002A402 [3.5/3.5]

201002A402 ..(KA's)

ANSWER: 031 (1.00)

c.

REFERENCE:

SSM 27-B, REV 7, P 4, LO #3

KA: 201002K402 [3.4/3.5]

201002K404 ..(KA's)

ANSWER: 032 (1.00)

c.

REFERENCE:

OTM SS282AR3, P 43 OF 65, LO #8

KA: 212000K412 [3.9/4.1]

212000K412 ..(KA's)

ANSWER: 033 (1.00)

c.

REFERENCE:

ISU: SSM 20I, p. 10, LO 3C.

KA: 262001A401 [3.4/3.7]

262001A401 ..(KA's)

ANSWER: 034 (1.00)

d.

REFERENCE:

SSM 20I, TABLE III, LO 8E.

KA: 262001K301 [3.5/3.7]

262001K301 ..(KA's)

ANSWER: 035 (1.00)

a.

REFERENCE:

SSM 12-A, p. 15, LO 4

KA: 204000K104 [2.9/2.9]

204000K104 ..(KA's)

ANSWER: 036 (1.00)

c.

REFERENCE:

SSM 15-e, p. 12, LO 6.

KA: 204000K404 [3.5/3.6]

204000K404 ..(KA's)

ANSWER: 037 (1.00)

a.

REFERENCE:

SSM 20-D, p. 29, LO 4a.

KA: 264000K402 [4.0/4.2]

264000K402 ..(KA's)

ANSWER: 038 (1.00)

d.

REFERENCE:

OTM SS282AR3, P 20 OF 65, LO #5

KA: 212000K502 [3.3/3.4]

212000K502 ..(KA's)

ANSWER: 039 (1.00)

b.

REFERENCE:

SSM 25D pg. 23. LO 9

K/A: 215005K505 [3.6/3.6]

215005K505 ..(KA's)

ANSWER: 040 (1.00)

d.

REFERENCE:

T.S. 3.6.2.1.a.2.c

K/A: 295026G004 [2.9/4.1]

295026G004 ..(KA's)

ANSWER: 041 (1.00)

d.

REFERENCE:

T.S. 4.4.1.1.2

K/A: 202001G005 [3.4/4.2]

202001G005 ..(KA's)

ANSWER: 042 (1.00)

a.

REFERENCE:

T.S. 3.6.1.6

K/A: 295028G004 [2.7/3.9]

295028G004 ..(KA's)

ANSWER: 043 (1.00)

a.

REFERENCE:

AOP 19.0, pg 3

K/A: 295018K202 [3.4/3.6]

295018K202 ..(KA's)

ANSWER: 044 (1.00)

c.

REFERENCE:

AOP 20.0, pg 4

K/A: 295019K214 [3.2/3.2]

295019K214 ..(KA's)

ANSWER: 045 (1.00)

c.

REFERENCE:

T.S. 3.8.2.1

K/A: 295003G004 [2.8/3.8]

295003G004 ..(KA's)

ANSWER: 046 (1.00)

a.

REFERENCE:

AOP-04.0 pg 4

K/A: 202002G014 [3.9/3.5]

202002G014 ..(KA's)

ANSWER: 047 (1.00)

c.

REFERENCE:

AOP-14.0, pg 3 and 4

K/A: 295010A206 [3.6/3.6]

295010A206 ..(KA's)

ANSWER: 048 (1.00)

c.

REFERENCE:

AOP-23.0, pg 3

K/A: 295009G011 [4.3/4.5]

295009G011 ..(KA's)

ANSWER: 049 (1.00)

b.

REFERENCE:

AOP-4.1, pg 3

K/A: 295014G010 [4.0/3.9]

295014G010 ..(KA's)

ANSWER: 050 (1.00)

d.

REFERENCE:

AOP - 16.0 Rev. 7, P. 3 OF 6

KA: 295006G010 [4.1/4.2]

295006G010 ..(KA's)

ANSWER: 051 (1.00)

a.

REFERENCE:

AOP-12.0, REV 4, P. 3 of 4

KA: 295004G011 [3.7/3.9]

295004G011 ..(KA's)

ANSWER: 052 (1.00)

c.

REFERENCE:

OPS-CLS-SM-300-C, REV 0, P 13 OF 35

KA: 295037G007 [3.5/3.7]

295037G007 ..(KA's)

ANSWER: 053 (1.00)

b.

REFERENCE:

AOP 04.4, Rev. 4, P. 3 of 4

KA: 295001A101 [3.5/3.6]

295001A101 ..(KA's)

ANSWER: 054 (1.00)

c.

REFERENCE:

SML LOI-CLS-SM-300-L, pg 26 and 102, LO 18

K/A: 295025G007 [3.5/3.7]

295025G007 ..(KA's)

ANSWER: 055 (1.00)

c.

REFERENCE:

T/S 3.4.3.2.c

KA: 295024G008 [3.6/4.4]

295024G008 ..(KA's)

ANSWER: 056 (1.00)

c.

REFERENCE:

AOP-18 section 3.1

KA: 295018G010 [3.4/3.3]

295018G010 ..(KA's)

ANSWER: 057 (1.00)

c.

REFERENCE:

EOP 2 & 3

KA: 295024G011 [3.9/3.7]

295024G011 ..(KA's)

ANSWER: 058 (1.00)

[illegible]. [+1.0]

REFERENCE:

Malfunction C&E #267, Rev 22, P. 1 of 3
SML-4-C 14-C, pg 33, LO 28e

KA: 295008A105 [3.3/3.3]

295008A105 ..(KA's)

ANSWER: 059 (1.00)

d.

REFERENCE:

Malfunction C&E #239, Rev 24, P. 1 of 2
KA: 295009A201 [4.2/4.2]
295009A201 ..(KA's)

ANSWER: 060 (1.00)

d.

REFERENCE:

Malfunction C&E #263, Rev 12, P. 1 of 2
SSM 14-2B, pg 22, LO 3j

KA: 206000K603 [2.9/3.1]

206000K603 ..(KA's)

ANSWER: 061 (1.00)

c.

REFERENCE:

BSEP/Vol. II/SD-46, Rev 10, P. 2

KA: 295019K303 [3.2/3.2]

295019K303 ..(KA's)

ANSWER: 062 (1.00)

d. [+1.0]

REFERENCE:

OPS-CLS-SM-300-D, pg 27, LO 1

K/A: 295031K205 [4.2/4.3]

295031K205 ..(KA's)

ANSWER: 063 (1.00)

a.

REFERENCE:

AOP-37.0, pg 4

K/A: 295002G010 [3.8/3.7]

295002G010 ..(KA's)

ANSWER: 064 (1.00)

a.

REFERENCE:

Malfunction C&E #324, Rev 21, P. 1 of 2
AOP-37.0, pg 3

KA: 295002K201 [3.5/3.5]

295002K201 ..(KA's)

ANSWER: 065 (1.00)

b.

REFERENCE:

Malfunction C&E #345, Rev 17, P. 2 of 5

KA: 295003A101 [3.7/3.8]

295003A101 ..(KA's)

ANSWER: 066 (1.00)

b.

REFERENCE:

SDM 15-e, p. 28

KA: 295021A206 [3.2/3.3]

295021A206 ..(KA's)

ANSWER: 067 (1.00)

d.

REFERENCE:

OPS-CLS-SM-300C, p. 5 & 30, LO 4, Reactor Scram Procedure.

KA: 295016G010 [3.8/3.6]

295016G010 ..(KA's)

ANSWER: 068 (1.00)

c.

REFERENCE:

AOP-30.0 page 5
OP-CLS-SM-300-A, LO 7

K/A: 295013G011 [4.1/4.4]

295013G011 ..(KA's)

ANSWER: 069 (1.00)

d.

REFERENCE:

T/S LSSS

KA: 295006G003 [3.8/4.4]

295006G003 ..(KA's)

ANSWER: 070 (1.00)

a.

REFERENCE:

EOP-01-UG p 56

KA: 295031G012 [3.9/4.5]

295031G012 ..(KA's)

ANSWER: 071 (1.00)

b.

REFERENCE:

SSM 18-2A, p. 26-27, L.O. 7.

K/A: 245000A301 [3.6/3.6]

245000A301 ..(KA's)

ANSWER: 072 (1.00)

d.

REFERENCE:

LPS-CLS-SM-300J, p. 12-14.

KA: 295015K201 [3.8/3.9]

295015K201 ..(KA's)

ANSWER: 073 (1.00)

b.

REFERENCE:

PEP-02.1 Sec. 4.2.2.
KA: 295038G011 [4.2*/4.5*]
295038G011 ..(KA's)

ANSWER: 074 (1.00)

d.

REFERENCE:

AOP 32,p. 3.

K/A: 295016G010 [3.8/3.6]

295016G010 ..(KA's)

ANSWER: 075 (1.00)

c.

REFERENCE:

AOP-15.0, "Alternate Shutdown Cooling Methods", Rev. 5, Page 7

KA: 295021A201 [3.5/3.6]

295021A201 ..(KA's)

ANSWER: 076 (1.00)

b.

REFERENCE:

LOI-CLS-SM-300-M, p. 17, LO 11.

KA: 295033A103 [3.8/3.8]

295033A103 ..(KA's)

ANSWER: 077 (1.00)

b.

REFERENCE:

LOI-CLS-SM-300-M, p. 25, "Secondary Containment Control", L.O. - 4.a

KA: 295032K303 [3.8/3.9]

295032K303 ..(KA's)

ANSWER: 078 (1.00)

a.

REFERENCE:

BSEP/Vol. II/SD-17, Rev. 13, P. 12

KA: 295007K203 [3.1/3.2]

295007K203 ..(KA's)

ANSWER: 079 (1.00)

a.

REFERENCE:

SML-LOI-CLS-SM-300-L, pg 61, LO 3a

K/A: 295026K301 [3.8/4.1]

295026K301 ..(KA's)

ANSWER: 080 (1.00)

d.

REFERENCE:

SML-LOI-CLS-SM-300-L, pg 71, LO 16

K/A: 295028K101 [3.5/3.7]

295028K101 ..(KA's)

ANSWER: 081 (1.00)

a.

REFERENCE:

OPS-CLS-SSM-304-1, pg 18, LO 1c. [illegible]d d.

K/A: 295016K303 [3.5/3.7]

295016K303 ..(KA's)

ANSWER: 082 (1.00)

c.

REFERENCE:

PEP-02.2, Emergency Control - Notification of Unusual Event, Rev. 14, Section 2.4

K/A: 294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 083 (1.00)

c.

REFERENCE:

10CFR20

K/A: 294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 084 (1.00)

a.

REFERENCE:

PEP-02.4, Emergency Control - Site Area Emergency, Rev.14, Section 1.1.10

K/A: 294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 085 (1.00)

b.

REFERENCE:

Radiation Control and Protection Manual, Rev. 21, Section 7.1.4

K/A: 294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 086 (1.00)

b.

REFERENCE:

OE&RC-0250, Posting of Areas/Materials, Rev. 19, Section 10.6

K/A: 294001K103 [3.3/3.8]

294001K103 ..(KA's)

ANSWER: 087 (1.00)

a.

REFERENCE:

OI-13, Valve and Electrical Lineup Controls, Rev. 35, Section 4.0.1

K/A: 294001K101 [3.7/3.7]

294001K101 ..(KA's)

ANSWER: 088 (1.00)

c.

REFERENCE:

OI-13, Valve and Electrical Lineup Controls, Rev. 35, Section 4.2.1.B

K/A: 294001K101 [3.7/3.7]

294001K101 ..(KA's)

ANSWER: 089 (1.00)

d.

REFERENCE:

OI-16, LCOs on Safety Related Equipment, Rev. 7, Section 3.2.1.1

K/A: 294001A110 [3.6/4.2]

294001A110 ..(KA's)

ANSWER: 090 (1.00)

c.

REFERENCE:

AI-107, Instructions For Working in Hot Environments, Rev. 5, Section 7.4

K/A: 294001K108 [3.1/3.4]

294001K108 ..(KA's)

ANSWER: 091 (1.00)

b.

REFERENCE:

AI-117, Guidance for Troubleshooting Safety Related Equipment, Rev. 0, Section 4.1

K/A: 294001A112 [3.5/4.2]

294001A112 ..(KA's)

ANSWER: 092 (1.00)

d.

REFERENCE:

AI-59, Jumpering and Wire Removal, Rev. 21, Section 5.1.1

K/A: 294001K107 [3.3/3.6]

294001K107 ..(KA's)

ANSWER: 093 (1.00)

a.

REFERENCE:

Technical Specification 6.8

K/A: 294001A101 [2.9/3.4]

294001A101 ..(KA's)

ANSWER: 094 (1.00)

a.

REFERENCE:

Student Study Material, Emergency Plan, Rev. 0, page 60

K/A: 294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 095 (1.00)

b.

REFERENCE:

PEP-03.8.1, Evacuation, Rev. 3, Section 3.2.2

K/A: 294001A116 [2.9/4.7]

294001A116 ..(KA's)

ANSWER: 096 (1.00)

c.

REFERENCE:

OI-13, "Valve & Electrical Lineup Administrative Controls", Rev. 034, Page 7

K/A: 294001K104 [3.3/3.6]

294001K104 ..(KA's)

ANSWER: 097 (1.00)

a.

REFERENCE:

OI-05, "Annunciator Status", Rev. 020, pages 1, 2

K/A: 294001A113 [4.5/4.3]

294001A113 ..(KA's)

ANSWER: 098 (1.00)

b.

REFERENCE:

10CFR50.54 (x) and (y)
Licensed Operator Training 07-D, page 29, Obj. 4

K/A: 294001A111 [3.3/4.3]

294001A111 ..(KA's)

ANSWER: 099 (1.00)

d.

REFERENCE:

Technical Specification 3.6

K/A: 295033K204 [3.9/4.2]

295033K204 ..(KA's)

ANSWER: 100 (1.00)

a.

REFERENCE:

SD-12, Primary Containment Isolation system

KA: 295020A101 [3.6/3.6]

295020A101 ..(KA's)

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

A N S W E R K E Y

MULTIPLE CHOICE

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

ANSWER KEY

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

ANSWER KEY

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

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

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

Organized by Question Number

QUESTION	VALUE	REFERENCE
001	1.00	8000001
002	1.00	8000002
003	1.00	8000003
004	1.00	8000004
005	1.00	8000008
006	1.00	8000009
007	1.00	8000010
008	1.00	8000011
009	1.00	8000012
010	1.00	8000013
011	1.00	8000014
012	1.00	8000015
013	1.00	8000016
014	1.00	8000017
015	1.00	8000018
016	1.00	8000019
017	1.00	8000020
018	1.00	8000022
019	1.00	8000023
020	1.00	8000025
021	1.00	8000026
022	1.00	8000028
023	1.00	8000033
024	1.00	8000035
025	1.00	8000037
026	1.00	8000039
027	1.00	8000041
028	1.00	8000042
029	1.00	8000043
030	1.00	8000044
031	1.00	8000045
032	1.00	8000046
033	1.00	8000048
034	1.00	8000049
035	1.00	8000050
036	1.00	8000051
037	1.00	8000052
038	1.00	8000053
039	1.00	8000054
040	1.00	8000055
041	1.00	8000056
042	1.00	8000057
043	1.00	8000058
044	1.00	8000059
045	1.00	8000061
046	1.00	8000062
047	1.00	8000063
048	1.00	8000064
049	1.00	8000065

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

Organized by Question Number

QUESTION	VALUE	REFERENCE
050	1.00	8000066
051	1.00	8000067
052	1.00	8000069
053	1.00	8000070
054	1.00	8000071
055	1.00	8000073
056	1.00	8000074
057	1.00	8000075
058	1.00	8000076
059	1.00	8000077
060	1.00	8000078
061	1.00	8000080
062	1.00	8000081
063	1.00	8000082
064	1.00	8000083
065	1.00	8000084
066	1.00	8000085
067	1.00	8000086
068	1.00	8000088
069	1.00	8000089
070	1.00	8000091
071	1.00	8000092
072	1.00	8000093
073	1.00	8000094
074	1.00	8000095
075	1.00	8000098
076	1.00	8000100
077	1.00	8000101
078	1.00	8000102
079	1.00	8000103
080	1.00	8000104
081	1.00	8000105
082	1.00	8000106
083	1.00	8000107
084	1.00	8000108
085	1.00	8000109
086	1.00	8000110
087	1.00	8000111
088	1.00	8000112
089	1.00	8000113
090	1.00	8000114
091	1.00	8000115
092	1.00	8000116
093	1.00	8000117
094	1.00	8000118
095	1.00	8000119
096	1.00	8000120
097	1.00	8000121
098	1.00	8000122

S R O Exam BWR Reactor

Organized by Question Number

QUESTION	VALUE	REFERENCE
099	1.00	8000126
100	1.00	8000128

	100.00	

	100.00	

S R O Exam BWR Reactor

Organized by KA Group

PLANT WIDE GENERICS

QUESTION	VALUE	KA
093	1.00	294001A101
089	1.00	294001A110
098	1.00	294001A111
091	1.00	294001A112
097	1.00	294001A113
094	1.00	294001A116
095	1.00	294001A116
082	1.00	294001A116
084	1.00	294001A116
088	1.00	294001K101
087	1.00	294001K101
085	1.00	294001K103
086	1.00	294001K103
083	1.00	294001K103
096	1.00	294001K104
092	1.00	294001K107
090	1.00	294001K108
PWG Total	17.00	

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
046	1.00	202002G014
016	1.00	203000A304
017	1.00	203000K117
015	1.00	206000K403
060	1.00	206000K603
018	1.00	209001A209
019	1.00	209001K601
009	1.00	211000A402
010	1.00	211000G005
032	1.00	212000K412
038	1.00	212000K502
020	1.00	215004K602
022	1.00	215005K103
039	1.00	215005K505
014	1.00	217000G007
013	1.00	217000G010
012	1.00	218000K101
007	1.00	223002A102
023	1.00	259002A201
005	1.00	261000K601

S R O Exam BWR Reactor

Organized by KA Group

PLANT SYSTEMS

Group I

QUESTION	VALUE	KA
033	1.00	262001A401
034	1.00	262001K301
037	1.00	264000K402

PS-I Total	23.00	

Group II

QUESTION	VALUE	KA
001	1.00	201001A204
030	1.00	201002A402
031	1.00	201002K404
041	1.00	202001G005
035	1.00	204000K104
036	1.00	204000K404
021	1.00	215003A401
027	1.00	234000K502
071	1.00	245000A301
024	1.00	271000A301
025	1.00	272000K603
028	1.00	286000A301
029	1.00	290003A301

PS-II Total	13.00	

Group III

QUESTION	VALUE	KA
003	1.00	239001K127
002	1.00	239001K201
026	1.00	268000K104
006	1.00	288000K603

PS-III Total	4.00	

PS Total	40.00	

EMERGENCY PLANT EVOLUTIONS

Group I

S R O Exam BWR Reactor

Organized by KA Group

EMERGENCY PLANT EVOLUTIONS

Group I

QUESTION	VALUE	KA
065	1.00	295003A101
045	1.00	295003G004
069	1.00	295006G003
050	1.00	295006G010
011	1.00	295007A104
078	1.00	295007K203
059	1.00	295009A201
048	1.00	295009G011
004	1.00	295009K301
047	1.00	295010A206
068	1.00	295013G011
049	1.00	295014G010
072	1.00	295015K201
067	1.00	295016G010
074	1.00	295016G010
081	1.00	295016K303
055	1.00	295024G008
057	1.00	295024G011
054	1.00	295025G007
040	1.00	295026G004
079	1.00	295026K301
008	1.00	295030K103
070	1.00	295031G012
062	1.00	295031K205
052	1.00	295037G007
073	1.00	295038G011

EPE-I Total	26.00	

Group II

QUESTION	VALUE	KA
053	1.00	295001A101
063	1.00	295002G010
064	1.00	295002K201
051	1.00	295004G011
058	1.00	295008A105
056	1.00	295018G010
043	1.00	295018K202
044	1.00	295019K214
061	1.00	295019K303
100	1.00	295020A101
075	1.00	295021A201
066	1.00	295021A206

S R O Exam B W R Reactor

Organized by KA Group

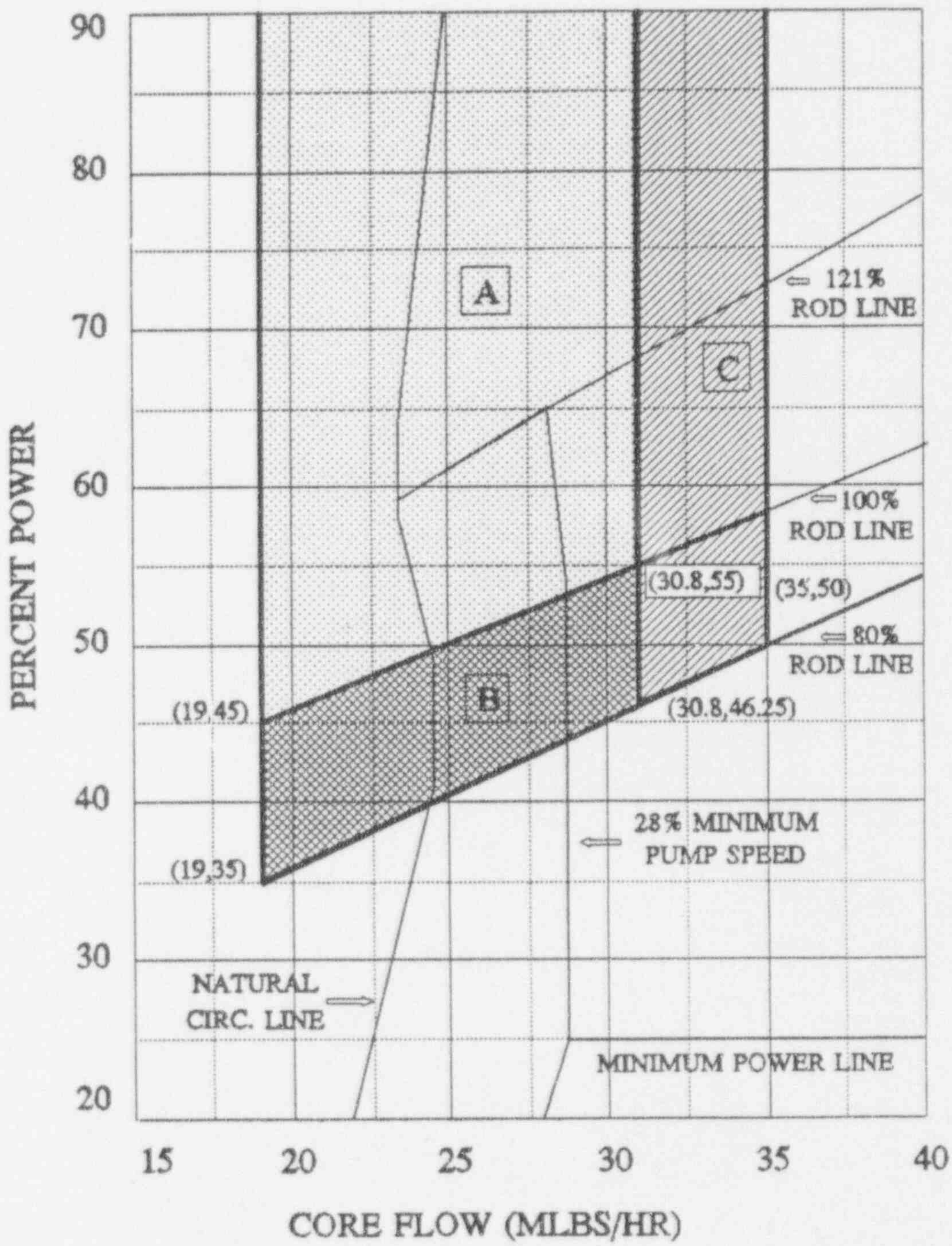
EMERGENCY PLANT EVOLUTIONS

Group II

QUESTION	VALUE	KA
042	1.00	295028G004
080	1.00	295028K101
077	1.00	295032K303
076	1.00	295033A103
099	1.00	295033K204
EPE-II Total	17.00	
EPE Total	43.00	
Test Total	100.00	

FIGURE 1

THERMAL POWER LIMITATIONS



CP&L

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK NUCLEAR PLANT

PLANT OPERATING MANUAL

VOLUME XIII

PLANT EMERGENCY PROCEDURE

UNIT
0

BNP RECIPIENT ID
FOR INFORMATION ONLY
Not to be used to perform maintenance, tests, surveillances, operate or manipulate plant systems, document activities, or write or implement other procedures.
073
CONTROLLED

0PEP-02.1

INITIAL EMERGENCY ACTIONS

REVISION 34

EFFECTIVE DATE
1/1/94

Sponsor Richard C Baldwin 12/20/93
Richard Baldwin Date

Approval Bradford L Houston 12/22/93
Manager - Emergency Preparedness Date

RECEIVED
DEC 29 1993
BNP DOC. CONTROL

LIST OF EFFECTIVE PAGES

PEP-02.1

TABLE OF CONTENTS

PEP-02.1 INITIAL EMERGENCY ACTIONS

1.0 PURPOSE

This procedure should be implemented by the Shift Supervisor or the individual listed in Step 3.0 upon recognition of an off-normal condition to assist in determining whether an event should be classified as an emergency.

2.0 REFERENCES

2.1 PEP-03.8.2, Personnel Accountability and Evacuation

2.2 PEP-03.9.2, First Aid and Medical Care

2.3 PEP-03.9.6, Search and Rescue

2.4 PEP-013, General Fire Plan

2.5 FEP-02.5, Emergency Control - General Emergency

2.6 PEP-02.4, Emergency Control - Site Area Emergency

2.7 PEP-02.3, Emergency Control - Alert

2.8 PEP-02.2, Emergency Control - Unusual Event

2.9 RCI-06.5, NRC Reporting Requirements

2.10 OI-51, NRC 1-Hour, 4-Hour, and 24-Hour Reporting Requirements

2.11 BSEP Technical Specification, Section 3.

2.12 OG-06.1, Radioactive Gaseous Release Control

2.13 BSP-44, Response to Personnel Injury and Accident Reporting

2.14 PFP-013, General Fire Plan (Vol. XIX, POM)

2.15 PEP-03.4.7, Automation of Off Site Dose Projections

3.0 RESPONSIBILITIES

3.1 The Shift Supervisor or alternate has immediate and unilateral authority to carry out this procedure. He may delegate specific steps as necessary, but shall not delegate the final classification decision.

3.2 The Senior Control Operator is a qualified alternate to implement this procedure if the Shift Supervisor, plant General Manager, or his designated alternate, are not available.

NOTE: Exhibit C at the end of this procedure provides a flowchart that addresses the SEC actions once an event has been declared.

4.0 DEFINITIONS/ABBREVIATIONS

None

5.0 GENERAL

Implementation of this procedure does not constitute an emergency but rather serves as a guideline for evaluation of the plant conditions and comparisons with Emergency Action Levels (EALs).

6.0 INITIATING CONDITION(S)

6.1 Once implemented, this procedure shall remain in effect until either:

6.1.1 The emergency is classified and the proper Emergency Control procedure is implemented

OR

6.1.2 The off-normal condition is resolved.

7.0 PRECAUTIONS/LIMITATIONS

As stated in the procedure.

8.0 SPECIAL TOOLS AND EQUIPMENT

None

9.0 PROCEDURE STEPS

NOTES:

1. If an action level for a higher classification was exceeded but the indicated level has currently abated or the situation has been resolved, then the higher classification should be reported to the state, counties, and NRC but should not be declared.

2. "*" denotes decisions or actions which should be entered in the Shift Foreman's Log.

3. The following actions are to be carried out by the Shift Supervisor (or his designated alternate) in an expeditious manner for personnel and plant protection and emergency classification.

9.1 Ensure appropriate Emergency Operating Procedures and plant procedures are being implemented.

9.0 PROCEDURE STEPS

9.2 Determine need to evacuate localized plant areas.

NOTE: If a Building Evacuation is not required, go to Step 9.3.

*9.2.1 Sound Building Evacuation alarm for 15 seconds and announce over the Plant PA System "(state emergency condition) in the (location). Evacuate the (location)."

Example: "Radiation Alarm in Radwaste Building, Evacuate the Radwaste Building."

NOTE: The TAC Building PA system can be used to notify personnel in the TAC Building by dialing 699-00 from any plant phone.

9.2.2 Implement Section 9.1 of PEP-03.8.2 and direct evacuees to report to their designated personnel assembly location and, direct work group supervisors to inform the Shift Supervisor of any personnel not accounted for within 30 minutes.

9.2.3 Repeat Step 9.2.1 above.

9.3 Determine whether personnel injuries have occurred.

NOTE: If no personnel injuries are reported, go to Step 9.4.

*9.3.1 Determine number of persons injured and their location(s).

9.3.2 Implement PEP-03.9.2 or PEP-03.9.6, as appropriate.

9.3.3 Determine whether injuries involve radioactive contamination.

NOTE: If contamination is involved, ensure appropriate precautions are taken in accordance with PEP 3.9:2.

CAUTION

Priority should be placed on lifesaving injury treatment over the need to decontaminate. See PEP-03.9.2 for guidance.

9.4 Determine whether off-normal conditions include fire.

NOTE: If no fire is detected or reported, go to Step 9.5.

9.4.1 Determine location of fire, sound fire alarm, and announce location using plant PA if not announced as part of Step 9.2.

*9.4.2 Implement PFP-013.

9.0 PROCEDURE STEPS

NOTE: For EAL flowpaths, the revision dates annotated in the top right corner, validated by dates in the List of Effective Pages, depict the form in use in lieu of the procedure revision level shown.

9.5 Using EAL flowpaths or Exhibit A, compare plant conditions (observed or indicated parameters and conditions) with the EALs and classify the emergency.

9.5.1 The EAL flowpath can be entered at any point if the event is known. (Example: fuel handling accident.) This point should be noted to ensure that all other events are evaluated prior to exiting the flowpath. If the event is not known, enter at Point A.

9.5.2 If no emergency exists (i.e., no emergency action level is met) go to Step 9.6.

9.5.3 If, at any time, a General Emergency declaration is warranted, the Site Emergency Coordinator is to immediately declare a General Emergency and carry out the appropriate actions. Implement PEP-02.5. Exhibit C can be used as a guide.

9.5.4 If an event (other than a General Emergency) is warranted, the Site Emergency Coordinator is to continue through the EAL flowpath after noting the level. The highest level will be declared upon completion of the flowpath.

9.5.5 If an EAL for a Site Area Emergency is met, implement PEP-02.4.

9.5.6 If an EAL for an Alert is met, implement PEP-02.3.

9.5.7 If an EAL for a Notification of Unusual Event is met, implement PEP-02.2.

9.6 Continue to monitor and evaluate plant conditions in accordance with previous steps until off-normal conditions are returned to normal.

9.7 Review RCI-06.5 and OI-51 to determine reporting requirements.

9.8 A checklist may be used to ensure that all essential tasks are completed; however, such a checklist shall not be used to replace this procedure (see Exhibit B).

NOTE: When operations are within normal operating parameters and safe in the judgment of the Shift Supervisor, terminate use of this procedure.

10.0 DIAGRAMS/ATTACHMENTS

10.1 Exhibit A, Emergency Action Levels

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

1.0 Abnormal Primary Leak Rate

1.1 Notification of Unusual Event

Reactor Coolant System total leakage greater than 25 gpm averaged over any 24-hour period using the sum of drywell equipment drain integrator (G16-FQ-K603) and drywell floor drain integrator (G16-FQ-K601), and the leakage rate has not been reduced to less than 25 gpm within eight hours, or plant shutdown is not achieved within required time period.

Unidentified Reactor Coolant System leakage greater than 5 gpm averaged over any 24-hour period using the drywell floor drain integrator (G16-FQ-K601), and the leakage rate has not been reduced to less than 5 gpm within eight hours, or plant shutdown is not achieved within required time period.

1.2 Alert

NOTE: See Exhibit A Section 2.4.1 if this occurs in conjunction with fuel damage or containment failure.

Small break LOCA with primary system leakage greater than 50 gpm. A LOCA is indicated by a significant loss of reactor inventory to the drywell resulting in increased drywell pressure, temperature, and/or sump pump usage indicated by:

- Low or falling Reactor Coolant System pressure with rising drywell pressure and temperature (C32-R608, CAC-PIC-2685, CAC-TR-4426-1A, CAC-TR-4426-1B, CAC-TR-4426-2A and CAC-TR-4426-2B).

1.3 Site Area Emergency

- Loss of coolant accident requiring the initiation of Low Pressure Coolant Injection, Core Spray, or the Automatic Depressurization System, AND REQUIRED FOR ADEQUATE CORE COOLING.

OR

- Loss of two-out-of-three fission product barriers listed in Exhibit A Section 2.4.1.

1.4 General Emergency

- Site Area Emergency indicated above AND inability to provide makeup water to the Reactor Coolant System (i.e., failure of HPCI, Core Spray A and B, RHR Loops A and B, RCIC, condensate, and feedwater) as indicated by falling or low reactor vessel level with attempts to inject water not successful.

OR

- Loss of two-out-of-three fission product barriers listed in Exhibit A Section 2.4.1 with a potential to lose the third barrier.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

2.0 Steam Line Break or Safety/Relief Valve Failure

2.1 Notification of Unusual Event

2.1.1 Reactor Coolant System pressure ≥ 1250 psig.

OR

2.1.2 Inability to close an SRV with Reactor Coolant System pressure ≤ 900 psig.

2.2 Alert

Steam line break downstream of MSIVs or upstream of feedwater isolation valves as indicated by:

A. Reactor trip with:

1. Low RCS pressure (C32-R608 or B21-PI-R605A or B21-PI-R605B)

OR

2. Low steam pressure (C32-R609)

OR

3. Low reactor vessel water level (C32-R608)

OR

4. High steam flow (C32-R603)

AND

B. Shift Supervisor/Site Emergency Coordinator's opinion or evidence on P601 and P603 of continuing steam flow with steam line break outside of primary containment.

2.3 Site Area Emergency

- Alert indicated above and inability to isolate the leak.

OR

- Loss of two-out-of-three fission product barriers listed in Exhibit A Section 2.4.1

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

2.0 <u>Steam Line Break or Safety/Relief Valve Failure</u>

2.4 General Emergency

2.4.1 Loss of any two of the three fission product barriers below with a potential loss of the third barrier.

A. Failed fuel causing RCS activity greater than 40 μCi/ml I-131 dose equivalent

B. Loss of primary coolant boundary

1. Loss of coolant accident (defined in Exhibit A Section 1.2 - Alert)

2. Major steam line break (defined in Exhibit A Section 2.2 - Alert)

<u>NOTE</u>: Primary containment integrity, according to technical specifications, shall exist when:

a. All penetrations required to be closed during accident conditions are either:

(1) Capable of being closed by an operable containment automatic isolation valve system, or;

(2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Technical Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each containment air lock is operable pursuant to Technical Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Technical Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is operable.

C. Loss of primary containment integrity as defined in technical specifications. A release path has been established.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

3.0 Abnormal Core Conditions and Core Damage

3.1 Notification of Unusual Event

Failed fuel as indicated by:

3.1.1 Liquid

A. Reactor Coolant System (RCS) activity greater than 4.0 μCi/ml I-131 dose equivalent

B. RCS activity greater than 0.2 μCi/ml I-131 dose equivalent but less than limit above for more than 48 hours

C. RCS activity greater than $100/\bar{E}$ μCi/ml for all isotopes

3.1.2 Gaseous

A. Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) reading of greater than 1.2×10^4 mR/hr

B. Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) increase of greater than 2.4×10^3 mR/hr in 30 minutes.

3.2 Alert

3.2.1 Liquid

NOTE: See Exhibit A Section 2.4.1 if this occurs in conjunction with a loss of primary coolant boundary or loss of primary containment integrity.

Reactor coolant activity greater than 40 μCi/ml I-131 dose equivalent

3.2.2 Gaseous

Steam jet air ejector off-gas radiation monitor (D12-RM-K601A and B) reading of greater than 1.2×10^5 mR/hr

3.3 Site Area Emergency

- Reactor Coolant System activity is greater than 400 μCi/ml I-131 dose equivalent.

OR

- Loss of two-out-of-three fission product barriers listed in Exhibit A Section 2.4.1.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

3.0 Abnormal Core Conditions and Core Damage

3.4 General Emergency

3.4.1 Any two functional high range drywell radiation monitors (D22-RI-4195, 4196, 4197, and 4198) reading greater than 5000 R/hr

OR

3.4.2 Reactor Coolant System activity is greater than 4000 μCi/ml I-131 dose equivalent

OR

3.4.3 Loss of two-out-of-three fission product barriers listed in Exhibit A Section 2.4.1 with a potential for loss of the third barrier.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

4.0 Abnormal Radiological Effluent or Radiation Levels

4.1 Notification of Unusual Event

4.1.1 Liquid Release

Any unplanned release from the liquid waste system resulting in activity levels in the discharge canal greater than those in 10CFR20, Appendix B, Table II, Column 2 (see Technical Specification 3.11.1.1).

4.1.2 Gaseous Release

Any gaseous release which exceeds the dose rate limit specified in Technical Specification 3.11.2.1(a) (i.e., exceeding the noble gas instantaneous dose rate limit as evaluated by OG-06.1).

4.1.3 Any building evacuation based on confirmed radiological conditions (i.e., greater than 10 mpc airborne [except precautionary evacuations]).

4.2 Alert

4.2.1 Liquid Release

Any liquid release resulting in activity concentration levels in the discharge canal that are greater than 10 times those given in 10CFR20, Appendix B, Table II, Column 2 (10 times the concentration listed in Unusual Event).

4.2.2 Gaseous Release

Any gaseous release which exceeds 10 times the dose rate limit specified in Technical Specification 3.11.2.1(a) (i.e., exceeding 10 times the noble gas instantaneous dose rate limit as evaluated by OG-06.1).

4.2.3 In-plant Leak or Spill

A. Any area radiation monitor or continuous air monitor off-scale high and radiological conditions are confirmed.

B. Any site evacuation based on confirmed radiological conditions.

C. Reactor Building closed cooling water monitor (D12-RM-K606) off-scale high and high activity is confirmed by sampling.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

4.0 <u>Abnormal Radiological Effluent or Radiation Levels</u>

4.3 Site Area Emergency

4.3.1 Projected dose exceeding 50 mRem Whole body (TEDE) <u>OR</u> exceeding 250 mRem Thyroid (CDE) at site boundary.

4.3.2 Measured dose rate exceeding 100 mR/hr at site boundary.

4.3.3 Measured I-131 equivalent concentration exceeds 3.9E-7 μCi/cc at the site boundary.

4.4 General Emergency

4.4.1 Offsite release resulting in a dose rate exceeding one (1) Rem Whole Body (TEDE) <u>OR</u> five (5) Rem Thyroid (CDE) at the Site Boundary as indicated by dose projection or field data.

4.4.2 Measured I-131 Equivalent concentration exceeding 3.9E-6 μCi/cc at a the site boundary.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

5.0 Loss of Shutdown Functions: Decay Heat and Reactivity

5.1 Notification of Unusual Event

N/A

5.2 Alert

5.2.1 Complete loss of ability to maintain plant in cold shutdown:

A. Loss of essential service water loops, or Loss of RHR loops A and B.

AND

B. Loss of Condenser Condensate System.

AND

C. Either:

1. Coolant temperature exceeds 212°F,

OR

2. Uncontrolled temperature rise approaching 212°F.

5.2.2 Failure of the Reactor Protection System to initiate and complete a scram, indicated on Panel A-5, which brings the reactor to a subcritical condition as indicated by full core display panel P603 and neutron monitoring instruments (APRM and IRM).

5.3 Site Area Emergency

Failure of the Reactor Protection System to initiate and complete a scram as indicated by Section 5.2.2 above.

AND

Failure of standby liquid control to bring the reactor to a subcritical condition.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

5.0 Loss of Shutdown Functions: Decay Heat and Reactivity

5.4 General Emergency

5.4.1 Site Area Emergency as indicated in Section 5.3 above lasting greater than 30 minutes.

AND

5.4.2 Loss of main condenser heat removal capability indicated by MSIVs shut or loss of vacuum on condenser vacuum indicator.

AND EITHER

A. Failure of all low pressure coolant injection trains indicated on panel P601.

OR

B. Failure of all service water trains necessary for decay heat removal indicated on panel P601 (RHR service water) and panel XU2 (nuclear and conventional service water).

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

6.0 <u>Electrical or Power Failures</u>

6.1 Notification of Unusual Event

6.1.1 Inability to power either 4 kV E Bus from off-site power.

<u>OR</u>

6.1.2 Loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize.

6.2 Alert

6.2.1 Loss of all vital DC power.

<u>OR</u>

6.2.2 A. Inability to power either 4 kV E Bus from off-site power.

<u>AND</u>

B. Loss of all on-site AC power capability indicated by failure of diesel generators to start or synchronize.

6.3 Site Area Emergency

Either Alert condition in Section 6.2.1 or 6.2.2 listed above <u>AND</u> lasting longer than 15 minutes.

6.4 General Emergency

N/A

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

7.0 Fire

7.1 Notification of Unusual Event

Fire within the protected area lasting longer than ten minutes.

7.2 Alert

Fire which could potentially affect vital safety-related equipment.

7.3 Site Area Emergency

Any fire that impairs the operability of any vital equipment which, in the opinion of the Site Emergency Coordinator, is essential to maintain the plant in a safe condition.

7.4 General Emergency

Any fire which in the opinion of the Site Emergency Coordinator could cause massive common damage to plant systems.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

8.0 Control Room Evacuation

8.1 Notification of Unusual Event

N/A

8.2 Alert

Evacuation of Control Room anticipated or required with control of shutdown established from local stations.

8.3 Site Area Emergency

Evacuation of Control Room AND local control of shutdown is not established in 15 minutes.

8.4 General Emergency

N/A

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

9.0 <u>Loss of Monitors or Alarms or Communication Capability</u>

9.1 Notification of Unusual Event

- Indications or alarms on process or effluent parameters not functional in the Control Room to an extent requiring plant shutdown or other significant loss of assessment or communication capability (see Section 9.5 Communication Failures Decision Matrix on page 18).

<u>OR</u>

- Loss of audible function on all alarms or annunciators for greater than 15 minutes.

9.2 Alert

Loss of audible and visual function on all alarms or annunciators for greater than 15 minutes.

9.3 Site Area Emergency

Loss of audible and visual function on all alarms or annunciators <u>AND</u> occurrence of a transient on the same unit.

9.4 General Emergency

N/A

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

9.0 Loss of Monitors or Alarms or Communication Capability

9.5 COMMUNICATION FAILURES DECISION MATRIX (DECLARATION OF A NOTIFICATION OF UNUSUAL EVENT)

		NOTIFICATION OF UNUSUAL EVENT
1.	Complete Loss of Selective Signaling	N
2.	Loss of NRC Emergency Notification System (ENS)	N
3.	Loss of Southern Bell Network	N
4.	Loss of CP&L Network (Microwave)	N
5.	Loss of Selective Signaling Phone and ENS	N
6.	Loss of Selective Signaling Phone and Southern Bell Network (Long Distance Calling)	N
7.	Loss of Selective Signaling Phone and CP&L Network (Microwave)	N
8.	Loss of ENS and Southern Bell Network (Microwave)	N
9.	Loss of ENS and CP&L Network (Microwave)	N
10.	Loss of BOTH Southern Bell and CP&L Network (Microwave)	Y
11.	Loss of Selective Signaling Phone, ENS, and Southern Bell Network (Long Distance Calling)	N
12.	Loss of Selective Signaling Phone, ENS, and CP&L Network (Microwave)	N
13.	Loss of All Phone Communication: Selective Signaling Phone, ENS, Southern Bell, [Long Distance Calling] and CP&L Network (See Note 1)	Y

NOTE:

1. See PEP-02.6.21 for alternate communication means.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

10.0 Fuel Handling Accident

10.1 Notification of Unusual Event

N/A

10.2 Alert

10.2.1 Fuel handling accident involving damage to new or spent fuel indicated by:

A. Observation/report AND alarm on:

1. Process Reactor Building ventilation RAD monitor D12-K609A, B or D12-RR-R605.

OR

2. Reactor Building roof ventilation monitor CAC-AIQ-1264-3.

OR

3. Refuel floor area monitor ARM channel 1-28 or 2-28.

10.3 Site Area Emergency

10.3.1 Major damage to spent fuel indicated by:

A. Observation of substantial damage to multiple fuel assemblies, or observation that water level has dropped below the top of the fuel.

AND

B. Indications or alarms listed in Section 10.2.1.A above.

10.4 General Emergency

N/A

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

11.0 Security Threats

11.1 Notification of Unusual Event

Declaration of a security alert as defined by the Security Contingency Plan.

11.2 Alert

Declaration of a security emergency as defined by the Security Contingency Plan.

11.3 Site Area Emergency

Physical attack on the plant involving imminent occupancy of the Control Room, auxiliary shutdown panels, and other vital areas.

11.4 General Emergency

Physical attack on the plant has resulted in unauthorized personnel occupying the Control Room and other vital areas.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

12.0 Injury or Specific LCOs

12.1 Notification of Unusual Event

CAUTION: If off-site medical assistance is to be requested to transport an injured individual to an off-site hospital, and if the injured is contaminated or the level of contamination cannot be determined, then the injured is considered a "contaminated injured individual."

12.1.1 Transportation of a contaminated or potentially contaminated injured individual from the site to an off-site hospital.

12.1.2 Emergency Core Cooling System initiated and discharging to vessel, other than by operator action, which in the opinion of the Shift Supervisor constitutes declaration of an unusual event.

12.1.3 Loss of containment integrity requiring shutdown by technical specifications and shutdown is not achieved within required time period.

12.1.4 Loss of engineered safety feature or Fire Protection System function requiring shutdown by technical specifications and shutdown is not achieved within required time period.

12.2 Alert

N/A

12.3 Site Area Emergency

N/A

12.4 General Emergency

N/A

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

13.0 Hazards to Plant Operations

13.1 Notification of Unusual Event

13.1.1 Aircraft crash within site boundaries with the potential to endanger safety-related equipment.

13.1.2 Unplanned explosion within the site boundaries with the potential to endanger safety-related equipment.

13.1.3 Release of toxic or flammable gas that could endanger personnel.

13.1.4 Turbine rotating component failure causing rapid plant shutdown.

13.2 Alert

13.2.1 Explosion, aircraft crash, or missile resulting in major damage to structures housing safety-related systems.

13.2.2 Unplanned and uncontrolled entry of toxic or flammable gases into vital areas in sufficient quantities to endanger personnel or the operability of safety-related equipment.

13.2.3 Turbine failure causing penetration of its outer casing.

13.3 Site Area Emergency

13.3.1 Explosion, aircraft crash, or missile resulting in major damage to safe shutdown equipment with plant not in cold shutdown.

13.3.2 Uncontrolled entry of flammable or toxic gases into vital areas where lack of access constitutes a safety problem with plant not in cold shutdown.

13.4 General Emergency

Any major internal or external event substantially beyond design basis which could cause massive common damage to plant systems.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

14.0 <u>Natural Events</u>

14.1 Notification of Unusual Event

14.1.1 Alarm on seismic monitor <u>AND</u> confirmation of earthquake.

14.1.2 Hurricane warning issued.

14.1.3 Tornado on site.

14.2 Alert

14.2.1 Earthquake registering greater than 0.08g on seismic instrumentation.

14.2.2 Any adverse weather conditions that causes a loss of function of two or more safety trains.

14.2.3 Tornado striking inside protected area resulting in major damage to structures housing safety-related systems.

14.2.4 Hurricane winds currently on site estimated by meteorological center to be:

A. 130 mph at 30 ft above ground level

B. 180 mph at 300 ft above ground level

14.3 Site Area Emergency

14.3.1 Earthquake registering greater than 0.16g on seismic instrumentation with plant not in cold shutdown.

14.3.2 Flood, low water, or hurricane surge greater than design levels or failure to protect vital equipment at lower levels and plant not in cold shutdown.

14.3.3 Plant not in cold shutdown with hurricane winds currently on site estimated by meteorological center to be:

A. 130 mph at 30 ft above ground level

B. 180 mph at 300 ft above ground level

14.4 General Emergency

Any major natural event substantially beyond design basis which could cause massive common damage to plant systems.

EXHIBIT A (Cont'd)

EMERGENCY ACTION LEVELS

15.0 Shift Supervisor/Site Emergency Coordinator Judgments

When any condition exists which indicates a necessity for an increased level of awareness or readiness above previous plant conditions, the Shift Supervisor/ Site Emergency Coordinator should use his judgment to declare the appropriate emergency status for the plant.

15.1 Notification of Unusual Event

Plant conditions exist that warrant increased awareness by plant staff such as exceeding any technical specification safety limit.

15.2 Alert

Plant conditions exist that reflect a significant degradation in the safety of the reactor, but releases from this event would be small.

15.3 Site Area Emergency

Plant conditions exist that involve major failures of equipment and that will lead to core damage. Unless corrective action is taken, significant radiation releases may occur.

15.4 General Emergency

Plant conditions exist that make a release of a large amount of radioactivity in a short time possible; any core melt situation.

EXHIBIT B

Site Emergency Coordinator-TSC
Turnover Checklist

Confirmed
(Init/Time)

I. Turnover Checklist - This checklist may be used as general guidance for turning over the Site Emergency Coordinator responsibilities from the Control Room to the Technical Support Center and for turning over off-site responsibilities from the Site Emergency Coordinator to the Emergency Response Manager.

A. SYNCHRONIZE-CLOCKS ____/____

B. ON-SITE SITUATION

1. Review Emergency Classification, basis for declaration, and mitigating actions ____/____

   a. Review status of safety equipment and system
   b. Review status of fission product barriers
   c. Review condition/stability of reactor
   d. Review any Emergency Action Levels exceeded
   e. Review cause, history, initiating events leading to declaration of emergency

2. Review on-site protective actions taken ____/____

   a. Assembly
   b. Shelter
   c. Evacuations (local, protected area, site exclusion area)
   d. Potassium iodide administration

3. Review state of off-site assistance request for the site ____/____

   a. Fire department
   b. Rescue squad
   c. Local law enforcement agency

C. OFF-SITE SITUATION

1. Review status of off-site notifications ____/____

   a. State and county initial and any follow-up messages
   b. NRC
   c. INPO
   d. Other: GE
   e. Any needed notifications that have not been made ____/____

EXHIBIT B (Cont'd)

Site Emergency Coordinator-TSC
Turnover Checklist (Cont'd)

	Confirmed (Init/Time)
2. Review protective action recommendations in notifications made to the state and counties.	/
3. Review any communications received from the state or counties regarding activation, readiness, protective actions, or requests for information	/
4. Review data on any projected or actual radiological releases	/
5. Review the time and content of any press releases or media briefings	/
D. EMERGENCY RESPONSE	/
1. Review status of Emergency Response Organization activation	/
a. Notifications made to off-duty and off-site personnel b. Emergency Response Facilities that are activated c. Emergency Response Facilities that will be activated d. Other notifications needed	
2. Review outside organizations requested to mobilize	/
3. Review assistance needed	/
4. Review personnel assigned to SERT and/or county EOCs	/
E. DECLARE ACTIVATION OF EMERGENCY RESPONSE FACILITY	/

EXHIBIT C (Page 1 of 3)

SITE EMERGENCY COORDINATOR
ACTIONS FLOW CHART

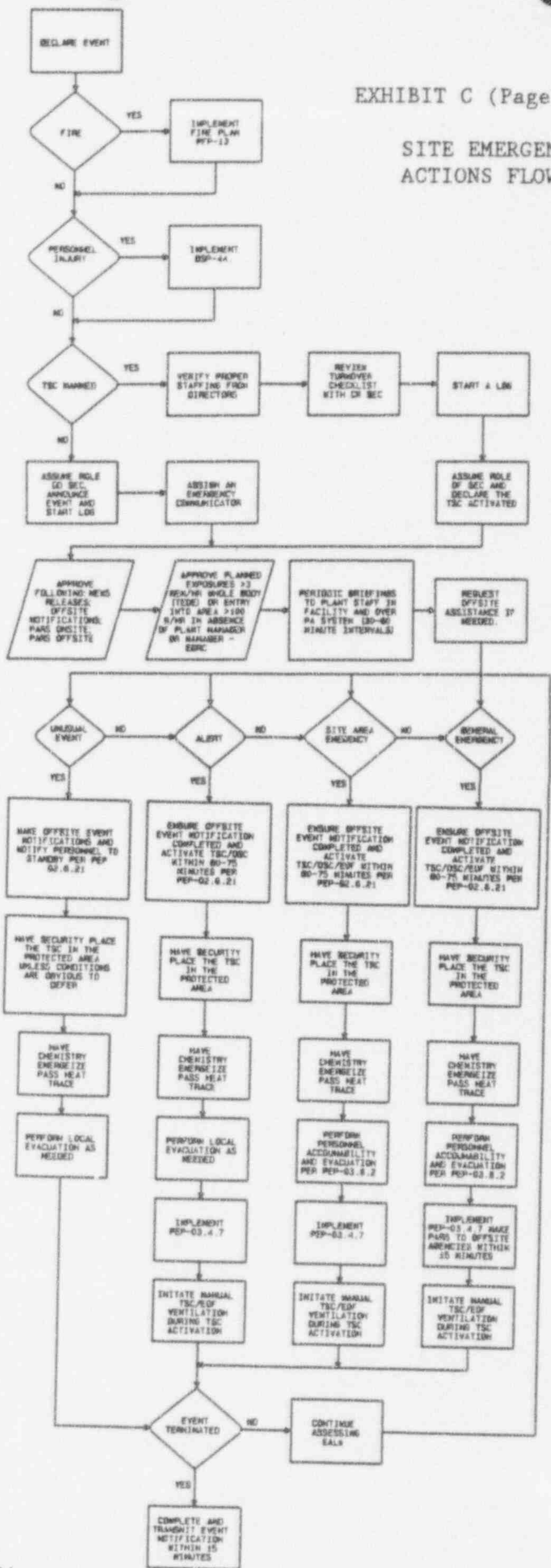
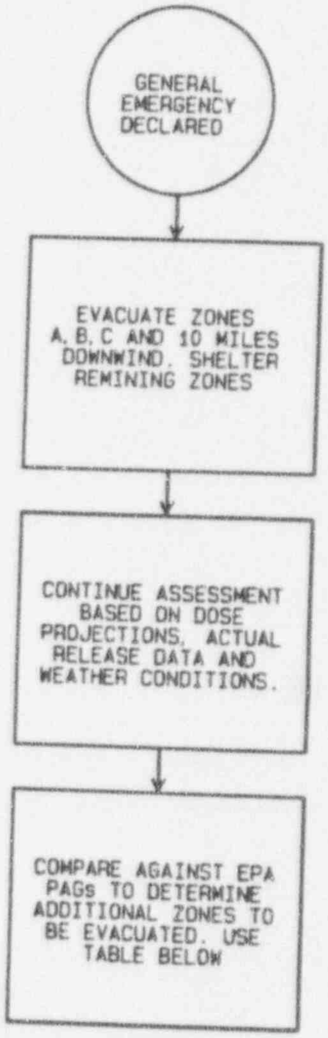


EXHIBIT C (Cont'd) (Page 2 of 3)

SITE EMERGENCY COORDINATOR

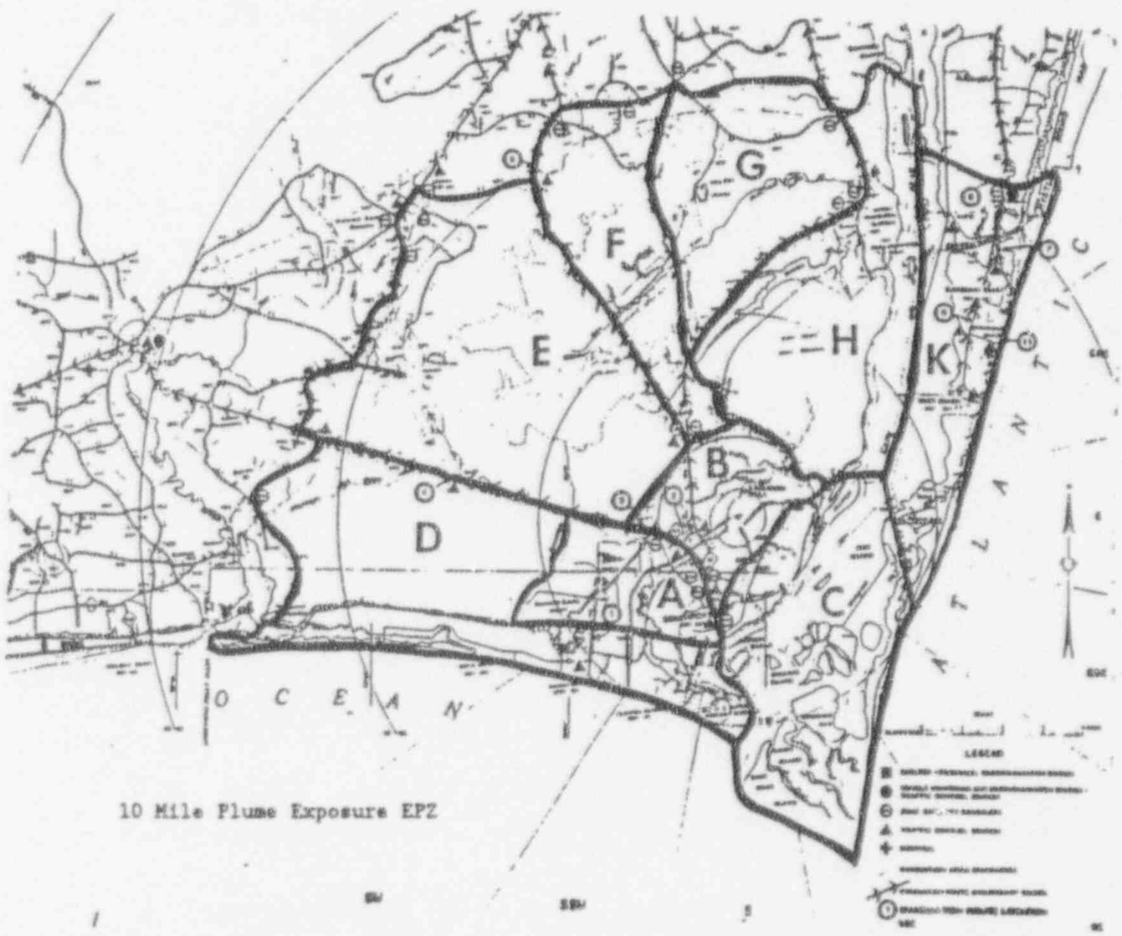


ACTIONS FLOW CHART

Projected Dose (Rem)	Recommended Action(s)	Comments
Whole Body (TEDE) < 1.0 or Thyroid (CDE) < 5.0	No planned protective action. State may issue an advisory to seek shelter and await further instructions. Monitor environmental radiation levels.	Previously recommended protective action may be reconsidered or terminated.
Whole Body (TEDE) 1 or above or Thyroid (CDE) 5 or above'	Conduct mandatory evacuation unless constraints make it impractical. Monitor environmental radiation levels. Control access	If constraints exist. Special considerations should be given for evacuation of children and pregnant women. Seeking shelter would be an alternative if evacuation were not immediately possible because of inadequate lead time due to rapid passage of the plume (puff release).

EXHIBIT C (Cont'd) (Page 3 of 3)
SITE EMERGENCY COORDINATOR
ACTIONS FLOW CHART

WIND FROM	EVACUATE ZONES	SHELTER ZONES	EVACUATION TIMES (MINS) WINTER/SUMMER
NORTH (338 - 022)	A,B,C,D	E,F,G,H,K	185 to 480
NORTHEAST (023 - 067)	A,B,C,D	E,F,G,H,K	185 to 480
EAST (068 - 112)	A,B,C,D,E	F,G,H,K	185 to 490
SOUTHEAST (113 - 157)	A,B,C,D,E,F,G,H	K	185 to 500
SOUTH (158 - 202)	A,B,C,E,F,G,H,K	D	185 to 685
SOUTHWEST (203 - 247)	A,B,C,G,H,K	D,E,F	175 to 685
WEST (248 - 292)	A,B,C,H,K	D,E,F,G,	175 to 685
NORTHWEST (293 - 337)	A,B,C,K	D,E,F,G,H,	175 to 685
ALL ZONES IN 10 MILE EPZ	A,B,C,D,E,F,G,H, K,		190 to 695



10 Mile Plume Exposure EPZ

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% Δk/k.

APPLICABILITY: CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% Δk/k:

a. Perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected, or

b. Be in HOT SHUTDOWN within 12 hours. Submit a Special Test Program to the Commission describing the methods to be used to determine the cause and magnitude of the reactivity difference.

SURVEILLANCE REQUIREMENTS

4.1.2 The ROD DENSITY shall be predicted and compared to the actual ROD DENSITY for selected operating conditions:

a. During the first start-up following CORE ALTERATIONS, and

b. At least once per effective full power month during POWER OPERATION.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within one hour:

      a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

      b) Disarm the associated directional control valves hydraulically by closing the insert and withdraw isolation valves.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within one hour:

      a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable control rods by at least two control cells in all directions, and

      b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*, or

      c) Fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves either:

         1) Electrically, or

         2) Hydraulically by closing the drive water and exhaust water isolation valves.

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

RETYPED TECH. SPECS.
Updated Thru. Amend.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. If the inoperable control rod(s) is inserted:

   a) Within one hour disarm the associated directional control valves either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

   b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE at least once per 31 days by:*

a. Verifying each valve to be open.

b. Cycling each valve at least one complete cycle of full travel.

4.1.3.1.2 All withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days when above the preset power level of the RWM and

b. At least once per 24 hours when above the preset power level of the RWM and any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE, when the reactor protection system logic is tested per Specification 4.3.1.2, by verifying that the drain and vent valves:

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open when the scram signal is reset or the scram discharge volume trip is bypassed.

*These valves may be closed intermittently for testing under administrative control.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 6, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that:

a. The control rod with the slow insertion time is declared inoperable,

b. The requirements of Specification 3.1.3.1 are satisfied, and

c. If within the preset power level of the RWM, the requirements of Specification 3.1.4.1.d are also satisfied, and

d. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 92 days when operation is continued with three or more control rods with slow scram insertion times;

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITIONS FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
46	0.358
36	1.096
26	1.860
6	3.419

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
46	0.379
36	1.162
26	1.971
6	3.624

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits, operation may continue and the provisions of Specification 3.0.4 are not applicable provided:

a. The control rods with the slower than average scram insertion times are declared inoperable,

b. The requirements of Specification 3.1.3.1 are satisfied, and

c. If within the preset power level of the RWM, the requirements of Specification 3.1.4.1.d are also satisfied, and

d. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 92 days when operation is continued with three or more control rods with slow scram insertion times.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one control rod scram accumulator inoperable, the provisions of Specification 3.0.4 are not applicable and operation may continue, provided that within 8 hours:

   1. The inoperable accumulator is restored to OPERABLE status, or

   2. The control rod associated with the inoperable accumulator is declared inoperable, and the requirements of Specification 3.1.3.1 are satisfied.

   3. And, if within the preset power level of the RWM, the requirements of Specification 3.1.4.1.d are also satisfied.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod scram accumulator inoperable, fully insert the affected control rod and electrically disarm the directional control valves within one hour. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The control rod scram accumulators shall be determined OPERABLE:

a. At least once per 7 days by verifying that the pressure and leak detectors are not in the alarmed condition, and

b. At least once per 18 months by performance of a:

   1. CHANNEL FUNCTIONAL TEST of the leak detectors, and

   2. CHANNEL CALIBRATION of the pressure detectors.

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: CONDITIONS 1, 2, and 5*.

ACTION:

a. In CONDITION 1 or 2 with one control rod not coupled to its associated drive mechanism, the provisions of Specification 3.0.4 are not applicable and operation may continue provided:

   1. Within the preset power level of the RWM, the control rod is declared inoperable and fully inserted until recoupling can be attempted with THERMAL POWER above the preset power level of the RWM and the requirements of Specification 3.1.4.1.d are satisfied.

   2. Above the preset power level of the RWM, the control rod drive is inserted to accomplish recoupling. If recoupling is not accomplished on the first attempt, declare the control rod inoperable, fully insert the control rod, and electrically disarm the directional control valves.

   3. The requirements of Specification 3.1.3.1 are satisfied.

b. In CONDITION 5*, with a withdrawn control rod not coupled to its associated drive mechanism, insert the control rod to accomplish recoupling. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The coupling integrity of a control rod shall be demonstrated by withdrawing the control rod to the fully withdrawn position and verifying that the rod does not go to the overtravel position:

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 All control rod reed switch position indicators shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2, and 5*.

ACTION:

a. In CONDITION 1 or 2: With one or more control rod reed switch position indicators inoperable, including "Full-in" or "Full-out" indication, the provisions of Specification 3.0.4 are not applicable and operation may continue, provided that within one hour:

   1) The position of the control rod is determined by an alternate method, or

   2) The control rod is moved to a position with an OPERABLE reed switch position indicator, or

   3) The control rod with the inoperable reed switch position indicator is declared inoperable and the requirements of Specification 3.1.3.1 are satisfied;

   4) And, if within the preset power level of the RWM, the requirements of Specification 3.1.4.1.d are also satisfied;

   Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In CONDITION 5* with a withdrawn control rod reed switch position indicator inoperable, fully insert the withdrawn control rod. The provisions of Specification 3.0.3 are not applicable.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod reed switch position indicators shall be determined OPERABLE by verifying:

a. At least once per 24 hours, that the position of the control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod reed switch position indicator corresponds to the control rod position indicated by the "Full-out" reed switches when performing Surveillance Requirement 4.1.3.6.b.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place when there is fuel in the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be inspected after reassembly and verified to be in place prior to start-up any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3/4 1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod Worth Minimizer (RWM) shall be OPERABLE when THERMAL POWER is less than 10% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*.

ACTION:

a. With the RWM inoperable after the first 12 control rods have been fully withdrawn on a startup, operation may continue provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

b. With the RWM inoperable before the first 12 control rods are withdrawn on a startup, one startup per calender year may be performed provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

c. With RWM inoperable on a shutdown, shutdown may continue provided that control rod movement and compliance with the prescribed BPWS control rod pattern are verified by a second licensed operator or qualified member of the plant technical staff.

d. With RWM operable but individual control rod(s) declared inoperable, operation and control rod movement below the preset power level of the RWM may continue provided:

    1. No more than three (3) control rods are declared inoperable in any one BWS group, and,

    2. The inoperable control rod(s) is bypassed on the RWM and control rod movement of the bypassed rod(s) is verified by a second licensed operator or qualified member of the plant technical staff.

e. With RWM inoperable, the provisions of Specification 3.0.4 are not applicable.

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

SURVEILLANCE REQUIREMENTS

4.1.4.1.1 The RWM shall be demonstrated OPERABLE in OPERATIONAL CONDITION 2, prior to withdrawal of control rods for the purpose of making the reactor critical and in OPERATIONAL CONDITION 1 when the RWM is initiated during control rod insertion when reducing THERMAL POWER by:

a. Verifying proper annunciation of the selection error of at least one out-of-sequence control rod, and

b. Verifying the rod block function of the RWM by moving an out-of-sequence control rod.

4.1.4.1.2 The RWM shall be demonstrated OPERABLE by verifying the control rod Banked Position Withdrawal Sequence input to the RWM computer is correct following any loading of the sequence program into the computer.

ROD SEQUENCE CONTROL SYSTEM

Pages 3/4 1-15 through 3/4 1-16 have been deleted.

(Next page is 3/4 1-17)

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 with:

a. THERMAL POWER greater than 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER and the MINIMUM CRITICAL POWER RATIO (MCPR) less than 1.70, or

b. THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER and the MCPR less than 1.40.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that either:

   1. The inoperable RBM channel is restored to OPERABLE status within 24 hours, or

   2. The redundant RBM is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable RBM is restored to OPERABLE status within 7 days.

   Otherwise, trip at least one rod block monitor channel.

b. With both RBM channels inoperable, trip at least one rod block monitor channel within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and during the OPERATIONAL CONDITIONS specified in Table 4.3.4-1.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE with:

a. An OPERABLE flow path from the storage tank to the reactor core, containing two pumps and two inline explosive injection valves,

b. The contained solution volume-concentration within the limits of Figure 3.1.5-1, and

c. The solution temperature above the limit of Figure 3.1.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that:

1. The volume and temperature of the sodium pentaborate solution are within the limits of Figures 3.1.5-1 and 3.1.5-2, and

2. The heating tracing circuit is OPERABLE.

b. At least once per 31 days by:

1. Starting each pump and recirculating demineralized water to the test tank,

2. Verifying the continuity of the explosive charge, and

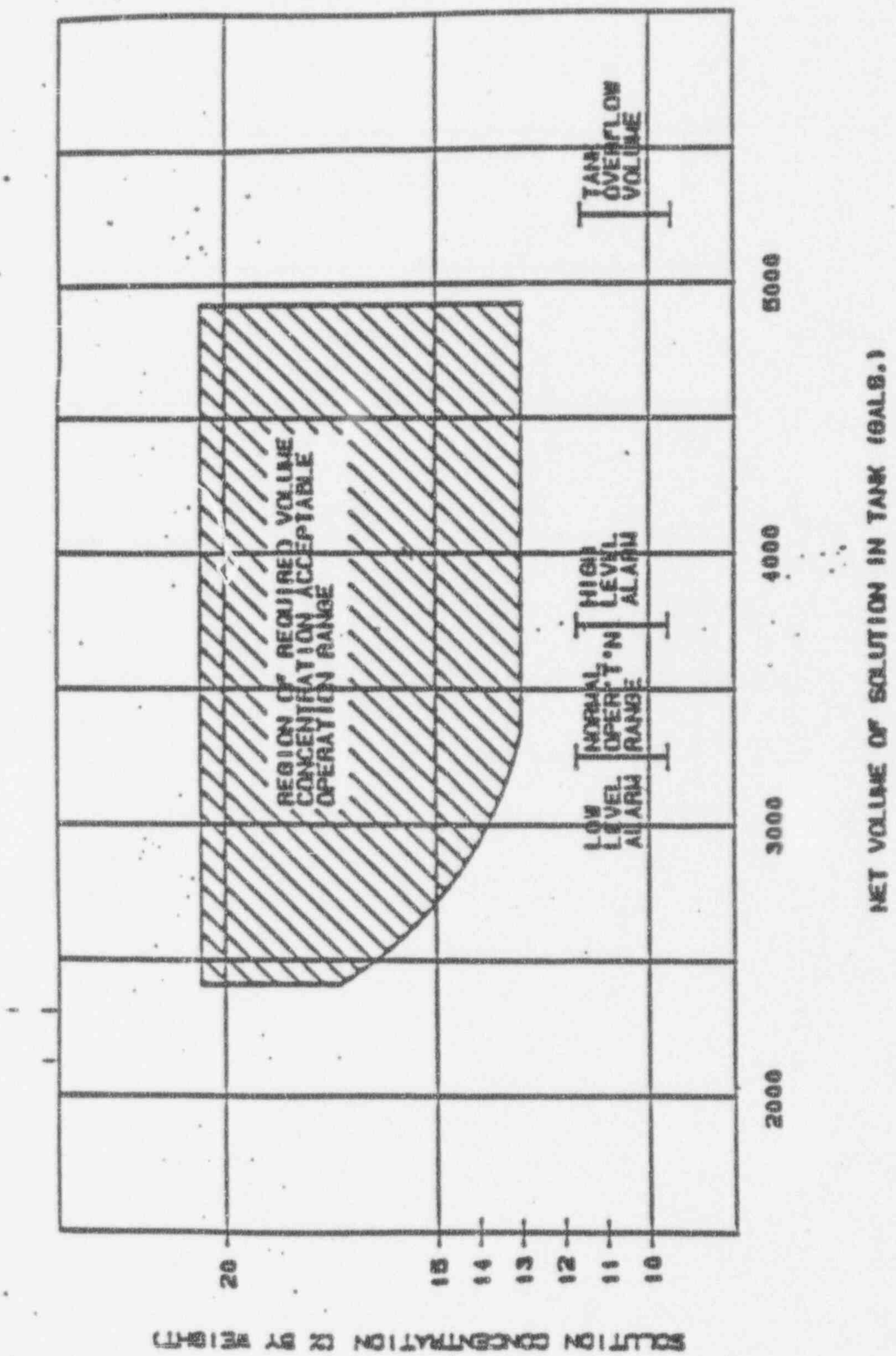
3. Determining the concentration of boron in solution by chemical analysis. This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit established in Figure 3.1.5-2.

c. At least once per 18 months during shutdown by:

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection test loops shall be tested in 36 months.

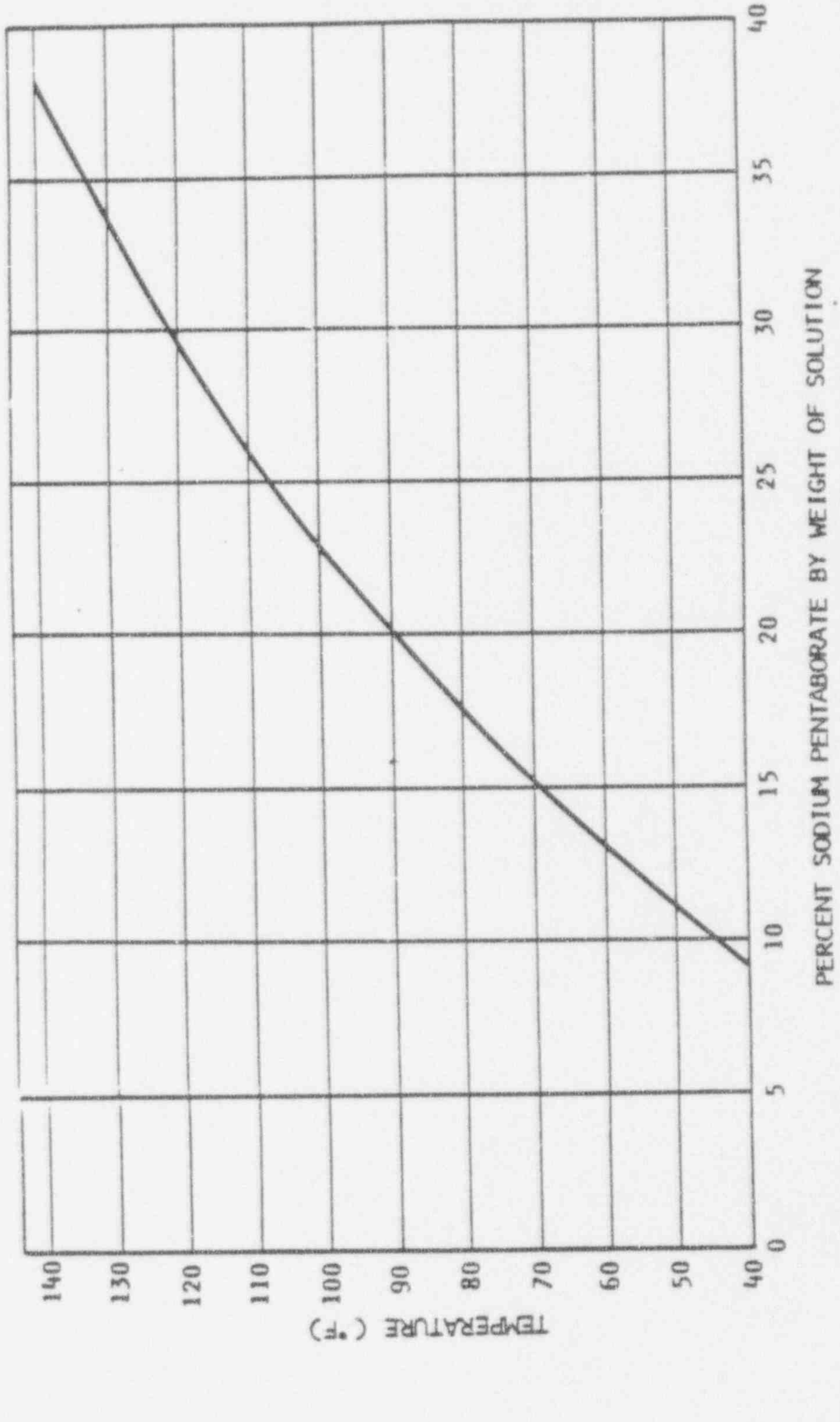
2. Demonstrating that the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1190 psig is met.

3. Demonstrating that the pump relief valve setpoint is 1450 ± 50 psig.



SODIUM PENTABORATE SOLUTION VOLUME CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1



SATURATION TEMPERATURE OF SODIUM
PENTABORATE SOLUTION

FIGURE 3.1.5-2

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 During power operation, the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall not exceed limits based on applicable APLHGR limit values that have been approved for the respective fuel and lattice type and determined by the approved methodology described in GESTAR-II. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value, adjusted for core flow and core power, for the most limiting lattice (excluding natural uranium) of each type of fuel shown in the applicable figures in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in Technical Specification 3.2.1 initiate corrective action within 15 minutes and continue corrective action so that APLHGR is within the required limits within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in Specification 3.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.2.2 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.2.1 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, core power, and cycle average exposure, shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT. The MCPR limits for ODYN OPTION A and ODYN OPTION B analyses, used in the above determination, shall be specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% RATED THERMAL POWER

ACTION:

With MCPR, as a function of core flow, core power, and cycle average exposure, less than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the applicable limit within 4 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2.1 MCPR, as a function of core flow, core power, and cycle average exposure, shall be determined to be equal to or greater than the applicable limit determined of Specification 3.2.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating in a LIMITING CONTROL ROD PATTERN for MCPR.

POWER DISTRIBUTION LIMITS

3/4.2.2 MINIMUM CRITICAL POWER RATIO (ODYN OPTION B)

LIMITING CONDITION FOR OPERATION

3.2.2.2 For the OPTION B MCPR limits provided in the CORE OPERATING LIMITS REPORT to be used, the cycle average 20% (notch 36) scram time (τ_{ave}) shall be less than or equal to the OPTION B scram time limit (τ_B), where τ_{ave} and τ_B are determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}, \text{ where}$$

i = Surveillance test number,
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test, and
τ_i = Average scram time to notch 36 for surveillance test i

$$\tau_B = \mu + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} (\sigma), \text{ where:}$$

i = Surveillance test number
n = Number of surveillance tests performed to date in the cycle (including BOC),
N_i = Number of rods tested in the i^{th} surveillance test
N_1 = Number of rods tested at BOC,
μ = 0.830 seconds
(mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to dropout on notch 36),
σ = 0.019 seconds
(standard deviation of the above statistical distribution)

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% RATED THERMAL POWER.

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION:

Within twelve hours after determining that τ_{ave} is greater than τ_B, the operating limit MCPRs shall be either:

a. Adjusted for each fuel type such that the operating limit MCPR is the maximum of the non-pressurization transient MCPR operating limit specified in the CORE OPERATING LIMITS REPORT or the adjusted pressurization transient MCPR operating limits, where the adjustment is made by:

$$MCPR_{adjusted} = MCPR_{option\ B} + \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}(MCPR_{option\ A} - MCPR_{option\ B})$$

where: τ_A = 1.096 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,

$MCPR_{option\ A}$ = Specified in the CORE OPERATING LIMITS REPORT,
$MCPR_{option\ B}$ = Specified in the CORE OPERATING LIMITS REPORT, or,

b. The OPTION A MCPR limits specified in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.2.2 The values of τ_{ave} and τ_B shall be determined and compared each time a scram time test is performed. The requirement for the frequency of scram time testing shall be identical to Specification 4.1.3.2.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 HIGH PRESSURE COOLANT INJECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.1 The High Pressure Coolant Injection (HPCI) system shall be OPERABLE with:

a. One OPERABLE high pressure coolant injection pump, and

b. An OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the pressure vessel.

APPLICABILITY: CONDITIONS 1, 2, and 3 with reactor vessel steam dome pressure greater than 150 psig.

ACTION:

a. With the HPCI system inoperable, POWER OPERATION may continue provided the ADS, CSS, and LPCI systems are OPERABLE; restore the inoperable HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the surveillance requirements of Specification 4.5.1 not performed at the required frequencies due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 48 hours after reactor steam pressure is adequate to perform the tests.

SURVEILLANCE REQUIREMENTS

4.5.1 The HPCI shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days, by verifying that the system develops a flow of at least 4250 gpm for a system head corresponding to a reactor pressure $\geq$ 1000 psig when steam is being supplied to the turbine at 1000, +20, -18, psig.

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

2. Verifying that the system develops a flow of at least 4250 gpm for a system head corresponding to a reactor pressure of $\geq$ 165 psig when steam is being supplied to the turbine at 165, $\pm$ 15, psig.

3. Verifying that the suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal or suppression pool high water level signal.

3/4.5.2 AUTOMATIC DEPRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.2 The Automatic Depressurization System (ADS) shall be OPERABLE with at least seven OPERABLE ADS valves.

APPLICABILITY: CONDITIONS 1, 2, and 3 with reactor vessel steam dome pressure > 113 psig.

ACTION:

a. With one ADS valve inoperable, POWER OPERATION may continue provided the HPCI, CSS, and LPCI systems are OPERABLE; restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two or more ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. With the Surveillance Requirement of Specification 4.5.2.b not performed at the required interval due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 12 hours after reactor vessel steam pressure is adequate to perform the tests.

SURVEILLANCE REQUIREMENTS

4.5.2 The ADS shall be demonstrated OPERABLE at least once per 18 months by:

a. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

b. Manually opening each ADS valve when the reactor steam dome pressure is $\geq$ 100 psig and observing that either;

1. The control valve or bypass valve position responds accordingly, or

2. There is a corresponding change in the measured steam flow.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 LOW PRESSURE COOLING SYSTEMS

CORE SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.3.1 Two independent Core Spray System (CSS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One pump, and

b. An OPERABLE flow path capable of taking suction from at least one of the following OPERABLE sources and transferring the water through the spray sparger to the reactor vessel:

   1. In OPERATIONAL CONDITION 1, 2, or 3, from the suppression pool.

   2. In OPERATIONAL CONDITION 4 or 5*:

      a) From the suppression pool, or

      b) When the suppression pool is inoperable, from the condensate storage tank containing at least 150,000 gallons of water.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

   1. With one CSS subsystem inoperable, POWER OPERATION may continue provided both LPCI subsystems are OPERABLE; restore the inoperable CSS subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

---

* The core spray system is not required to be OPERABLE provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

3. In the event the CSS is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

b. In OPERATIONAL CONDITION 4 or 5*:

1. With one CSS subsystem inoperable, operation may continue provided that at least one LPCI subsystem is OPERABLE within 4 hours; otherwise, suspend all operations that have a potential for draining the reactor vessel.

2. With both CSS subsystems inoperable, operation may continue provided that at least one LPCI subsystem is OPERABLE and both LPCI subsystems are OPERABLE within 4 hours. Otherwise, suspend all operations that have a potential for draining the reactor vessel and verify that at least one LPCI subsystem is OPERABLE within 4 hours.

3. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.5.3.1 Each CSS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the condensate storage tank minimum required volume when the condensate storage tank is required to be OPERABLE.

b. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.

* The core spray system is not required to be OPERABLE provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 92 days by:

1. Verifying that each CSS pump can be started from the control room and develops a flow of at least 4625 gpm on recirculation flow against a system head corresponding to a reactor vessel pressure of $\geq$ 113 psig.

2. Performing a CHANNEL CALIBRATION of the core spray header ΔP instrumentation and verifying the setpoint to be 5, $\pm$1.5, psid greater than the normal indicated ΔP.

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

LIMITING CONDITION FOR OPERATION

3.5.3.2 Two independent Low Pressure Coolant Injection (LPCI) subsystems of the residual heat removal system shall be OPERABLE with each subsystem comprised of:

a. Two pumps,

b. An OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: CONDITIONS 1, 2, 3, 4*, and 5*.

ACTION:

a. In CONDITION 1, 2, or 3:

1. With one LPCI subsystem or one LPCI pump inoperable, POWER OPERATION may continue provided both CSS subsystems are OPERABLE; restore the inoperable LPCI subsystem or pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both LPCI subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the LPCI system cross-tie valve open or power not removed from the valve operator, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

b. In CONDITION 4* or 5* with one or more LPCI subsystems inoperable, take the ACTION required by Specification 3.5.3.1. The provisions of Specification 3.0.3 are not applicable.

*Not applicable when two CSS subsystems are OPERABLE per Specification 3.5.3.1.

cyb 6/11/93
~~See TSI 87-02~~

SURVEILLANCE REQUIREMENTS

4.5.3.2 Each LPCI subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water,

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

3. Verifying that the subsystem cross-tie valve is closed with power removed from the valve operator.

b. At least once per 92 days by verifying each pair of LPCI pumps discharging to a common header can be started from the control room and develops a total flow of at least 17,000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 20 psig.

c. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel is excluded from this test.

3/4.5.4 SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.5.4 The suppression pool shall be OPERABLE with a minimum water level $\geq$-31 inches except the suppression pool may be inoperable:

a. In OPERATIONAL CONDITION 4, provided that:

1. The reactor mode switch is locked in the Shutdown position, and

2. The core spray system is OPERABLE per Specification 3.5.3.1 with an OPERABLE flow path capable of taking suction from the OPERABLE condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

b. In OPERATIONAL CONDITION 5, provided that:

1. The reactor mode switch is locked in the Refuel position, and

2. The core spray system is OPERABLE per Specification 3.5.3.1 with an OPERABLE flow path capable of taking suction from the OPERABLE condensate storage tank, and transferring the water through the spray sparger to the reactor vessel, or

3. The reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

APPLICABILITY: CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

a. In CONDITIONS 1, 2, or 3 with the water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In CONDITIONS 4 or 5, with the suppression pool inoperable and the above conditions not satisfied, suspend all operations in the reactor vessel, all positive reactivity changes, and all operations that have a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The suppression pool shall be determined OPERABLE by verifying the water level to be within the limit at least once per 12 hours.

4.5.4.2 The above conditions shall be verified to be satisfied prior to making the suppression pool inoperable and at least once per 12 hours thereafter until the suppression pool is restored to OPERABLE status.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all primary containment penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 or Specification 3.6.3.

b. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

c. By verifying the suppression pool OPERABLE per Specification 3.6.2.1.

* Except valves, blind flanges, and deact[illegible]ted automatic valves which are located inside the containment, the MSI[illegible]it, the RWCU Penetration Triangle Room, or the TIP Room, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days. Those valves located above the drywell head requiring head shield block removal for verification will be verified prior to each replacement of the shield blocks.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 49 psig, or

2. Less than or equal to L_t, 0.357 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, subject to Type B and C tests when pressurized to P_a, 49 psig.

c. *Less than or equal to 11.5 scf per hour for any one main steam line isolation valve when tested at 25 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line isolation valve,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, subject to Type B and C tests to less than or equal to 0.60 L_a, and

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* Exemption to Appendix "J" of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve,

prior to increasing reactor coolant system temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following schedule and shall be determined in conformance with the criteria specified in Appendix J of 10CFR50:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 49 psig, or P_t, 25 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a or 0.75 L_t, a Type A test shall be performed at each plant shutdown for refueling or every 18 months, whichever occurs first, until two consecutive Type A tests meet 0.75 L_a or 0.75 L_t, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a or 0.25 L_t.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 49 psig or P_t, 25 psig.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 49 psig, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Main steam line isolation valves.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

h. The provisions of Specification 4.0.2 are not applicable to 24 month and 40 ± 10 month surveillance intervals.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 49 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock door interlock inoperable:

1. Lock the inner air lock door closed.

2. Operation may then continue provided that the inner air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

c. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying the seal leakage rate to be less than or equal to 5 scf per hour when the gap between the door seals is pressurized to 10 psig*:

1. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at lease once per 72 hours, and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used and no maintenance has been performed on the air lock, and

3. When the air lock seal has been replaced.

b. By conducting an overall air lock leakage that at P_a, 49 psig, and by verifying that the overall air lock leakage is within its limit:

1. At least once per six months#, and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance (except for seal replacement) has been performed on the air lock that could affect the air lock sealing capability.*

c. By verification of air lock interlock OPERABILITY:

1. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used, and

2. Prior to and following a drywell entry when PRIMARY CONTAINMENT INTEGRITY is required, and

3. Following the performance of maintenance affecting the air lock interlock.

* Exemption of Appendix J of 10 CFR 50.

\# The provisions of Specification 4.0.2 are not applicable

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.4.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2. This Special Report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

PRIMARY CONTAINMENT INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment internal pressure shall be maintained between -0.5 and 1.75 psig.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

With the containment internal pressure outside of the specified limit, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Primary containment average air temperature shall not exceed 135°F.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the primary containment average air temperature > 135°F*, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The primary containment average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. Below 5' elevation,

b. Between 10' and 23' elevation,

c. Between 28' and 45' elevation,

d. Between 70' and 80' elevation, and

e. Above 90' elevation.

* The primary containment average air temperature limit may be increased to 140°F until August 15, 1985, at which time the limit will be returned to 135°F.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 87,600 ft^3 and 89,600 ft^3, equivalent to a level between -27 inches and -31 inches, and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

c) 120°F with the main steam line isolation valves closed following a scram.

b. Two OPERABLE suppression chamber water temperature instrumentation channels with a minimum of 11 operable RTD inputs per channel.

c. A total leakage from the drywell to the suppression chamber of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 1 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION: (Continued)

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F manually scram the reactor and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With one suppression chamber water temperature instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify suppression chamber water temperature to be within the limits at least once per 12 hours.

d. With both suppression chamber water temperature instrumentation channels inoperable, restore at least one inoperable temperature instrumentation channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

   1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature to be less than or equal to 105°F.

   2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

      a) Suppression chamber average water temperature to be less than or equal to 110°F, and

      b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER.

   3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

c. By an external visual examination of selected emergency core cooling system suction line penetrations of the suppression chamber enclosure prior to taking the reactor from COLD SHUTDOWN after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 160°F and reactor coolant system pressure greater than 200 psig.

d. By verifying at least two suppression chamber water temperature instrumentation channels OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours.

   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   3. CHANNEL CALIBRATION at least once per 18 months (550 days).

   with the temperature alarm setpoint for high water temperature less than or equal to 95°F.

e. At least once per 18 months by:

   1. A visual inspection of the accessible interior of the suppression chamber and exterior of the suppression chamber enclosure.

SURVEILLANCE REQUIREMENTS (Continued)

2. Conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 1 psig and verifying that the differential pressure does not decrease by more than 0.25 inches of water per minute for a 10 minute period.

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent cooling loops, each loop consisting of two pumps and one heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one RHR suppression pool cooling loop inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable; restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both RHR suppression pool cooling loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days by verifying that each RHR pump can be started from the control room and develops a flow of at least 7,700 gpm on recirculation flow through the RHR heat exchanger and the suppression pool.

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~~See TSI 87-02~~

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves specified in Table 3.6.3-1 shall be OPERABLE with isolation times as shown in Table 3.6.3-1, and the reactor instrumentation system isolation valves shall be OPERABLE.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves specified in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that at least one isolation valve is maintained OPERABLE in each affected penetration that is open and either:

   1. The inoperable valve(s) is restored to OPERABLE status within 8 hours, or

   2. Each affected penetration line is isolated within 8 hours by use of at least one deactivated automatic valve secured in the isolation position, or

   3. Each affected penetration line is isolated within 8 hours by use of at least one closed manual valve or blind flange.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation system isolation valves inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 8 hours;

   1. The inoperable valve is returned to OPERABLE status, or

   2. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve specified in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of the cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve specified in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation system isolation valve shall be demonstrated OPERABLE at least once per 18 months by cycling each valve through at least one complete cycle of full travel.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

Table 3.6.3-1 has been deleted.

Refer to Plant Procedure RCI-02.6.

Pages 3/4 6-15 through 3/4 6-17 have been deleted.

(Next page is 3/4 6-18)

CONTAINMENT SYSTEMS

3/4.6.4 VACUUM RELIEF

DRYWELL - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 All drywell-suppression chamber vacuum breakers shall be OPERABLE and in the closed position with:

a. The position indicator OPERABLE, and

b. An opening setpoint of less than or equal to 0.5 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With no more than 2 drywell-suppression chamber vacuum breakers inoperable for opening but known to be in the closed position, the provisions of Specification 3.0.4 are not applicable and operation may continue until the next COLD SHUTDOWN provided the surveillance requirements of Specification 4.6.4.1.a are performed on the OPERABLE vacuum breakers within 4 hours and at least once per 15 days thereafter until the inoperable vacuum breakers are restored to OPERABLE status; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one drywell-suppression chamber vacuum breaker in the open position, as indicated by the position indicating system, the provisions of Specification 3.0.4 are not applicable and operation may continue provided the surveillance requirements of Specification 4.6.4.1.a are performed on the OPERABLE vacuum breakers and the surveillance requirements of Specification 4.6.4.1.b are performed within 8 hours and at least once per 72 hours thereafter until the inoperable vacuum breaker is restored to the closed position; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the position indicator of any drywell-suppression chamber vacuum breaker inoperable, the provisions of Specification 3.0.4 are not applicable and operation may continue, provided the surveillance requirements of Specification 4.6.4.1.b are performed within 8 hours and at least once per 72 hours thereafter until the inoperable position indicator is returned to OPERABLE status; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each drywell-suppression chamber vacuum breaker shall be demonstrated OPERABLE:

a. At least once per 31 days and after any discharge of steam to the suppression chamber from any source, by exercising each vacuum breaker through one complete cycle and verifying that each vacuum breaker is closed as indicated by the position indication system.

b. Whenever a vacuum breaker is in the open position, as indicated by the position indication system, by conducting a test that verifies that the differential pressure is maintained greater than 1/2 the initial delta P for one hour without N_2 makeup.

c. At least once per 18 months during shutdown by:

   1. Verifying the opening setpoint, from the closed position, to be less than or equal to 0.5 psid,

   2. Performance of a CHANNEL CALIBRATION that each position indicator indicates the vacuum breaker to be open if the vacuum breaker does not satisfy the delta P test in 4.6.4.1.b.

SUPPRESSION POOL - REACTOR BUILDING VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 All suppression pool-Reactor Building vacuum breakers shall be OPERABLE with:

a. an opening setpoint of less than or equal to 0.5 psid

b. an OPERABLE Nitrogen Backup System consisting of two independent subsystems (one subsystem for each vacuum breaker).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool-Reactor Building vacuum breaker inoperable for opening but known to be in the closed position, restore the inoperable vacuum breaker to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one Nitrogen Backup System subsystem inoperable, verify the remaining subsystem is OPERABLE and restore the inoperable subsystem to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With both Nitrogen Backup System subsystems inoperable, restore at least one inoperable subsystem to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2.1 Each suppression pool-Reactor Building vacuum breaker shall be demonstrated OPERABLE:

a. At least once per 92 days by:

    1. Manually verifying that each vacuum breaker check valve is free to open, and

    2. Cycling each vacuum breaker butterfly valve through at least one complete cycle of full travel.

b. At least once per 18 months by:

    1. Demonstrating that the force required to open each vacuum breaker check valve does not exceed 0.5 psid.

2. Demonstrating that the vacuum breaker butterfly valve opens at -0.45 ± 0.05 psid, drywell pressure going negative relative to Reactor Building pressure.

3. Visual inspections.

4.6.4.2.2 The Nitrogen Backup System shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that each subsystem is pressurized to greater than or equal to 1130 psig.

b. At least once per 18 months by verifying that each subsystem maintains system pressure with a leakage rate of less than or equal to .65 SCFM at a starting pressure greater than or equal to 1130 psig.

c. At least once per 18 months by performing a logic system functional test to ensure actuation of the nitrogen backup system.

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: CONDITIONS 1, 2, 3, 5, and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY, restore SECONDARY CONTAINMENT INTEGRITY within 8 hours, or:

a. In CONDITION 1, 2, OR 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In CONDITION 5 or *, suspend irradiated fuel handling in the secondary containment, CORE ALTERATIONS, and activities which could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by verifying:

a. At least once per 92 days that each secondary containment isolation damper is OPERABLE or secured in the closed position per Specification 3.6.5.2.

b. At least once per 18 months by operating a standby gas treatment system for 1 hour and maintaining $\geq$ 1/4 inch of vacuum, water gauge, at a flow rate not exceeding 3000 CFM.

*When irradiated fuel is being handled in the secondary containment.

RETYPED TECH. SPECS.
Updated Thru. Amend.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5, and *.

ACTION:

With one or more of the secondary containment isolation dampers specified in Table 3.6.5.2-1 inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable, provided that at least one isolation damper is maintained OPERABLE in each affected penetration that is open, and:

a. The inoperable damper is restored to OPERABLE status within 8 hours, or

b. The affected penetration is isolated by use of a closed damper within 8 hours, or

c. SECONDARY CONTAINMENT INTEGRITY is demonstrated within 8 hours and the damper is restored to OPERABLE status within 7 days.

Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL CONDITION 5 or *, suspend irradiated fuel handling in the secondary containment, CORE ALTERATIONS, or activities that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

*When irradiated fuel is being handled in the secondary containment.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment automatic isolation damper specified in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each automatic isolation damper testable during plant operation through at least one complete cycle of full travel.

b. Prior to returning the damper to service after maintenance, repair, or replacement work is performed on the damper or its associated actuator, control, or power circuit by performance of the cycling test and verification of isolation time.

c. At least once per 18 months during COLD SHUTDOWN or REFUELING by:

   1. Cycling each automatic damper through at least one complete cycle of full travel and measuring the isolation time, and

   2. Verifying that on a secondary containment isolation test signal each automatic damper actuates to its isolation position.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

Table 3.6.5.2-1 has been deleted.

Refer to Plant Procedure RCI-02.6.

CONTAINMENT SYSTEMS

3/4.6.6 CONTAINMENT ATMOSPHERE CONTROL

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5, and *.

ACTION:

a. With one standby gas treatment subsystem inoperable:

   1. In OPERATIONAL CONDITION 1, 2, or 3, restore the inoperable sybsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In OPERATIONAL CONDITION 5 or *, restore the inoperable subsystem to OPERABLE status within 31 days or suspend irradiated fuel handling in the secondary containment, CORE ALTERATIONS, or operations that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable;

   1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

   2. In OPERATIONAL CONDITION 5 or *, suspend all irradiated fuel handling in the secondary containment, CORE ALTERATIONS, or operations that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

   a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on automatic control.

*When irradiated fuel is being handled in the secondary containment.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 3000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

   3. Verifying a system flow rate of 3000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8.5 inches Water Gauge while operating the filter train at a flow rate of 3000 cfm ± 10%.

   2. Verifying that the filter train starts on each secondary containment isolation test signal.

   3. Verifying that the heaters will dissipate at least 15.2 kw when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3000 cfm $\pm$ 10%.

CONTAINMENT ATMOSPHERE DILUTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.2 The containment atmosphere dilution (CAD) system shall be OPERABLE with:

a. An OPERABLE flow path capable of supplying nitrogen to the drywell, and

b. A minimum supply of 4350 gallons of liquid nitrogen.

APPLICABILITY: CONDITION 1*.

ACTION:

With the CAD system inoperable, restore the CAD system to OPERABLE status within 31 days or be in at least STARTUP within the next 8 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The CAD system shall be demonstrated to be OPERABLE:

a. At least once per 31 days by verifying that:

1. The system contains a minimum of 4350 gallons of liquid nitrogen, and

2. Each valve (manual, power-operated, or automatic) in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months by:

1. Cycling each power-operated (excluding automatic) valve in the flow path through at least one complete cycle of full travel, and

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Group 2 and 6 isolation test signal.

*When oxygen concentration is required to be < 4% per Specification 3.6.6.3.

OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3* The primary containment atmosphere oxygen concentration shall be less than 4% by volume during the period from:

a. Within 24 hours after THERMAL POWER > 15% of RATED THERMAL POWER, to

b. Within 24 hours prior to a scheduled reduction of THERMAL POWER to < 15% of RATED THERMAL POWER.

APPLICABILITY: CONDITION 1.

ACTION:

With the oxygen concentration in the primary containment exceeding the limit, be in at least START-UP within 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The oxygen concentration in the primary containment shall be verified to be within the limit within 24 hours after THERMAL POWER > 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

*For the period commencing at 0630 on June 29, 1981, a temporary exemption is allowed to operate BSEP-2 in Condition 1 with containment oxygen concentration exceeding 4% by volume for 72 hours.

See TSI 87-03

CONTAINMENT SYSTEMS

GAS ANALYZER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.4 Two independent gas analyzer systems for the drywell and suppression chamber shall be OPERABLE with each system consisting of an oxygen analyzer and a hydrogen analyzer.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With one oxygen and/or one hydrogen analyzer inoperable, restore at least two oxygen and two hydrogen analyzers to OPERABLE status within 31 days or be in at least STARTUP within the next 8 hours. The provisions of Specification 3.0.4 are not applicable.

b. With no gas analyzer OPERABLE for oxygen and/or hydrogen, be in at least STARTUP within 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.4 Each gas analyzer system shall be demonstrated OPERABLE at least once per 92 days by performing a CHANNEL CALIBRATION using standard gas samples containing a nominal:

a. Zero volume percent hydrogen, balance nitrogen.

b. Seven to ten volume percent hydrogen, balance nitrogen.

c. Twenty-five to thirty volume percent hydrogen, balance nitrogen.

d. Zero volume percent oxygen, balance nitrogen.

e. Seven to ten percent oxygen, balance nitrogen.

f. Twenty to twenty-five percent oxygen, balance nitrogen.

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 Two independent Residual Heat Removal Service Water (RHRSW) System subsystems shall be OPERABLE with each subsystem comprised of:

a. Two pumps, and

b. An OPERABLE flow path for heat removal capable of taking suction from the intake canal via the service water system and transferring the water through an RHR heat exchanger.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

a. With one RHRSW pump inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable; restore the inoperable pump to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one RHRSW subsystem inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable; restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With both RHRSW subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each residual heat removal service water subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

b. At least once per 92 days by verifying that each pump develops a pump ΔP of at least 232 psi at a flow of 4000 gpm measured through the heat exchanger with a minimum suction pressure of 20 psig.

See TSI 89-02

RETYPED TECH. SPECS.
Updated Thru. Amend. 78

3/4.7.1 SERVICE WATER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.1.2 The Service Water System shall be OPERABLE with at least:

In OPERATIONAL CONDITIONS 1, 2, and 3:

Three OPERABLE site nuclear service water pumps, and two OPERABLE conventional service water pumps capable of supplying the nuclear and conventional headers.

In OPERATIONAL CONDITIONS 4 AND 5:

Three OPERABLE site nuclear service water pumps, and two OPERABLE Unit 2 service water pumps, nuclear and/or conventional, powered from separate emergency buses and capable of supplying the nuclear header.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2, or 3:

1. With one OPERABLE conventional service water pump:

    a. Ensure that, if only one Unit 2 nuclear service water pump is OPERABLE, the OPERABLE conventional service water pump is powered from a separate emergency bus than the OPERABLE Unit 2 nuclear service water pump, and

    b. Restore at least one additional conventional service water pump to OPERABLE status within 7 days.

    Otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With no conventional service water pumps OPERABLE:

    a. Ensure both Unit 2 nuclear service water pumps are OPERABLE, and

    b. Restore at least one conventional service water pump to OPERABLE status within 72 hours.

    Otherwise, be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

3. With two OPERABLE site nuclear service water pumps, unless the provisions of ACTION b.4 apply for Unit 1, restore one additional site nuclear service water pump within 7 days or be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

4. With two OPERABLE site nuclear service water pumps and one OPERABLE conventional service water pump:

   a. Ensure at least one Unit 2 nuclear service water pump is OPERABLE, and

   b. Ensure that, if only one Unit 2 nuclear service water pump is OPERABLE, the OPERABLE conventional service water pump is powered from a separate emergency bus than the OPERABLE Unit 2 nuclear service water pump, and

   c. Restore two conventional service water pumps or three site nuclear service water pumps to OPERABLE status within 72 hours.

   Otherwise, be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

5. With less than two OPERABLE site nuclear service water pumps, be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITIONS 4 or 5:

   1. With one OPERABLE Unit 2 service water pump, restore at least two Unit 2 service water pumps to OPERABLE status within 7 days. Otherwise, suspend all operations that have a potential for draining the reactor vessel.

   2. With no OPERABLE Unit 2 service water pumps, suspend all operations that have a potential for draining the reactor vessel.

   3. With two OPERABLE site nuclear service water pumps, unless the provisions of ACTION b.4 apply, restore at least one additional nuclear service water pump to OPERABLE status within 7 days. Otherwise, take the ACTION required by Specification 3.8.1.2.

   4. With the service water system nuclear header inoperable, operation of both units may continue provided that two Unit 1 nuclear service water pumps are OPERABLE, both units' nuclear service water header valves are administratively controlled as required to ensure cooling water to the diesel generators, at least two Unit 2 conventional service water pumps are OPERABLE on the conventional header, and vital ECCS loads are aligned to the conventional service water system header. Restore the service water system nuclear header and at least three site nuclear service water pumps to OPERABLE status within 14 days. Otherwise, take the ACTION required by Specification 3.8.1.2.

   5. With less than two OPERABLE site nuclear service water pumps, take the ACTION required by Specification 3.8.1.2.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The service water system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on the appropriate ECCS actuation test signals.

c. At least once per 92 days by verifying that the alternate diesel generator service water supply valve will open on a low header pressure signal.

3/4.7.2 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Control Room Emergency Ventilation System shall be OPERABLE with:

a. An OPERABLE Radiation/Smoke Protection Mode consisting of two OPERABLE control room emergency filtration subsystems.

b. An OPERABLE Chlorine Protection Mode.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5, *, and **

ACTION:

a. In OPERATIONAL CONDITIONS 1 and 2:

1. With one control room emergency filtration unit inoperable, restore the inoperable control room emergency filtration unit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both control room emergency filtration units inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 3:

1. With one control room emergency filtration unit inoperable, restore the inoperable control room emergency filtration unit to OPERABLE status within 7 days or be in COLD SHUTDOWN within the following 24 hours.

2. With both control room emergency filtration units inoperable, be in COLD SHUTDOWN within the following 24 hours.

c. In OPERATIONAL CONDITIONS 4, 5, and *:

1. With one control room emergency filtration unit inoperable, restore the inoperable control room emergency filtration unit to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control building emergency filtration unit in the Radiation/Smoke Protection Mode.

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* During movement of irradiated fuel assemblies in the secondary containment.

** The Chlorine Protection Mode is required to be OPERABLE at all times when the chlorine tank car is within the exclusion area.

3/4.7.2 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued):

2. With both control room emergency filtration units inoperable, suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in secondary containment, and operations with a potential for draining the reactor vessel.

d. With the Chlorine Protection Mode inoperable, within 8 hours remove the chlorine tank car from the exclusion area. If the tank car physically can not be removed from the exclusion area, take the ACTIONS required in items a.2, b.2, and c.2 above.

SURVEILLANCE REQUIREMENTS

4.7.2 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow, from the control room, through the HEPA filter and charcoal adsorbers in each filtration unit and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structual maintenance on the HEPA filter or charcoal adsorber housing, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria of > 99 percent efficiency using the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 2000 cfm ± 10%.

3/4.7.2 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

3. Verifying a system flow rate of 2000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 5.25 inches Water Gauge while operating the filter train at a flow rate of 2000 cfm ± 10%.

2. Verifying that on a smoke detector or control room ventilation system high radiation test signal, the control building ventilation system automatically diverts its inlet flow through the HEPA filters and charcoal adsorber banks of the emergency filtration system.

3. Verifying that on a chlorine detector test signal, the control building ventilation system automatically isolates and the control room emergency filtration system cannot be started by a smoke detector or control room ventilation system high radiation test signal.

4. Verifying that the system maintains the control room at a positive pressure relative to the outside atmosphere during system operation.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove > 99 percent of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove > 99 percent of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2,000 cfm ± 10%.

PLANT SYSTEMS

3/4.7.3 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.3 Flood protection shall be provided for all safety-related systems, components, and structures when the water level of the intake canal exceeds 17'6" Mean Sea Level USGS datum.

APPLICABILITY: At all times.

ACTION:

With the water level above elevation 17'6" Mean Sea Level USGS datum:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

b. Initiate the applicable emergency procedures to mitigate the consequences of flooding vital equipment.

SURVEILLANCE REQUIREMENTS

4.7.3 The water level of the intake canal shall be determined to be within the limit by:

a. Measurement at least once per 2 hours when the water level is equal to or above elevation 15'0" Mean Sea Level USGS datum,

b. Performing a CHANNEL FUNCTIONAL TEST of the high water level instrumentation at least once per 92 days, and

c. Performing a CHANNEL CALIBRATION of the high water level instrumentation at least once per 18 months.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 113 psig.

ACTION:

With the RCIC system inoperable, operation may continue and the provisions of Specifications 3.0.4 are not applicable provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 113 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the highpoint vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 400 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after reactor steam pressure is adequate to perform the test.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and restart+ and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

2. Verifying that the system will develop a flow of greater than or equal to 400 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 $\pm$ 15 psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.

* The provisions of Specifications 4.0.4 are not applicable provided the surveillance is performed within 24 hours after reactor steam pressure is adequate to perform the tests.

\+ Automatic restart on a low water level signal which is subsequent to a high water level trip.

3/4.7.5 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.5 All hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.5.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.5 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-1. The visual inspection interval for each type of snubber shall be determined base upon the criteria provided in Table 4.7-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 182.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual

SURVEILLANCE REQUIREMENTS (Continued)

inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.5.f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meeting the functional test acceptance criteria of Specification 4.7.5.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.5-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.5.f. The cumulative number of snubbers of a type tested is denoted by "N." At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.5-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during a day of equipment failure are retested.

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the operable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the operability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.5.e for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

TABLE 4.7-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

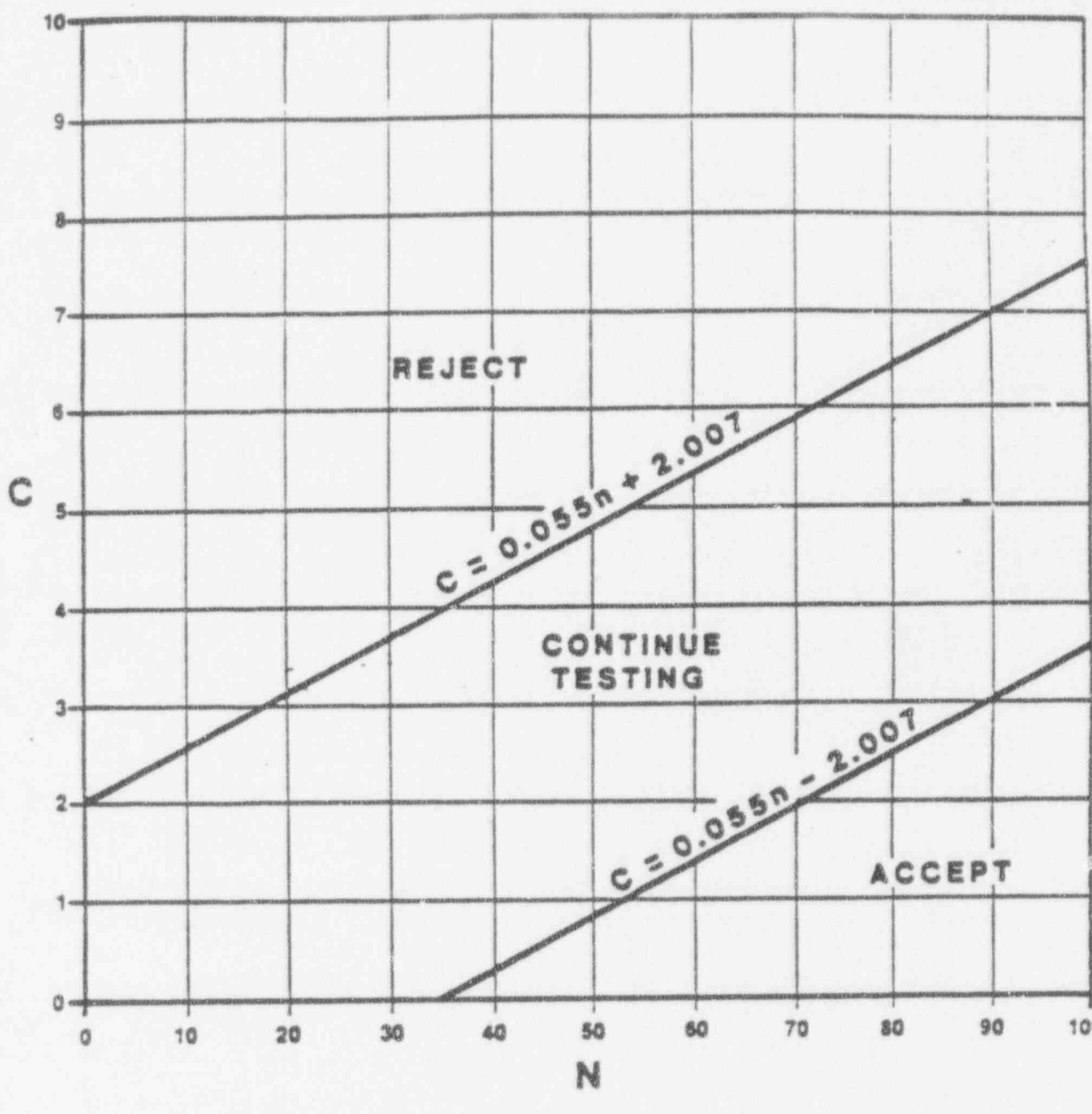
Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer included a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.



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BRUNSWICK - UNIT 2 Carolina Power & Light Company	Sample Plan 2) FOR SNUBBER FUNCTIONAL TEST Figure 4.7.5-1	Page 3/4 7-14

3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.6 Each sealed source containing radioactive material in excess of 100 microcuries of beta and/or gamma-emitting material or 5 microcuries of alpha-emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

a. Either decontaminated and repaired, or

b. Disposed of in accordance with Commission Regulations.

The Provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.6.2 Test Frequencies - Each category of sealed sources (excluding start-up sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed and fission detector source shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed in use.

c. Start-up sources and fission detectors - Each sealed start-up source and fission detector shall be tested within 31 days prior to being subjected to core flux and or installed in the core following repair or maintenance to the source.

4.7.6.3 Reports - A Special Report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

3/4.7.7 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The fire suppression water system shall be OPERABLE with:

a. Two OPERABLE fire pumps, one motor-driven and one diesel-driven, each with a capacity of 2000 gpm, with their discharges aligned to the fire suppression yard main,

b. The fire protection water tank, with a minimum contained volume of 200,000 gallons, and the demineralized water tank, with a minimum contained volume of 90,000 gallons, and

c. An OPERABLE flow path capable of taking suction from each of the water supplies and transferring the water through the yard main and distribution piping with OPERABLE sectionalizing control or isolation valves to, but not including, the yard hydrant curb valves and the first valve ahead of each sprinkler and hose standpipe system required to be OPERABLE per Specifications 3.7.7.2 and 3.7.7.4.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

  1. Establish a backup fire suppression water system within 24 hours, or

  2. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

  3. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2;

     a) By telephone within 24 hours,

     b) Confirmed by telegraph, mailgram, or facsimile transmission no later than the first working day following the event, and

     c) In writing within 14 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume is at least the minimum specified.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each pump and operating it for at least 15 minutes.

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each pump develops at least 2000 gpm at a system head of 125 psig,

   2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3. Verifying that each fire pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 125 psig.

f. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The fuel storage tank contains at least 500 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 20 minutes.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water, and sediment.

c. At least once per 18 months, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and

2. Verifying the diesel starts from ambient conditions on the auto-start signal and operates for greater than or equal to 20 minutes while loaded with the fire pump.

4.7.7.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.2 The following spray and/or sprinkler systems shall be OPERABLE:

a. Diesel Generator #1 Preaction System - Diesel Generator Building

b. Diesel Generator #2 Preaction System - Diesel Generator Building

c. Diesel Generator #3 Preaction System - Diesel Generator Building

d. Diesel Generator #4 Preaction System - Diesel Generator Building

e. South Cable Spread Area Sprinkler System - Diesel Generator Building

f. North Cable Spread Area Sprinkler System - Diesel Generator Building

g. Two Standby Gas Treatment Train 1A Deluge Systems - Reactor Building #2.

h. Two Standby Gas Treatment Train 1B Deluge Systems - Reactor Building #2.

i. Area Sprinkler System - Reactor Building #2.

j. Service Water Pump Area Sprinkler System - Service Water Building

k. Service Water Cable Spread Area Sprinkler System - Service Water Building

l. Drumming Room Sprinkler System - Radwaste Building

m. Makeup Water Treatment Area Sprinkler System - Makeup Water Treatment Building

APPLICABILITY: Whenever equipment in the areas protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation signal, and

   b) Cycling each valve in the flow path thr. is not testable during plant operation through at leas: .e complete cycle of full travel.

2. By inspection o. the spray .eaders to verify their integrity, and

3. By inspection of each deluge nozzle to verify no blockage.

PLANT SYSTEMS

HIGH PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.3 The following high pressure CO_2 systems shall be OPERABLE with a minimum contained weight of 67.5 lbs. of CO_2 in each cylinder of the inservice bank.

a. Unit No. 2 HPCI CO_2 System - Unit No. 2 Reactor Building.

b. Control Building CO_2 System - Control Building.

APPLICABILITY: Whenever equipment in the area protected by the high pressure CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required high pressure CO_2 systems inoperable, establish backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying that the high pressure CO_2 cylinders contain at least the minimum specified weight of CO_2.

b. At least once per 18 months by verifying:

1. The system control heads and associated ventilation dampers actuate manually and automatically, as appropriate, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through flooding system headers and nozzles and hose reel system hoses to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.7.4 The fire hose stations shown in Table 3.7.7.4-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7.7.4-1 inoperable, within one hour:

   1. Provide an alternative means of fire suppression for the unprotected area(s), or

   2. Route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.4 Each of the fire hose stations shown in Table 3.7.7.4-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

TABLE 3.7.7.4-1

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Unit No. 2 Reactor Bldg.	-17'	2-RB-19
	-17'	2-RB-20
	-17'	2-RB-24
	-17'	2-RB-25
	-17'	2-RB-26
	20'	2-RB-21
	20'	2-RB-22
	20'	2-RB-23
	20'	2-RB-27
	20'	2-RB-28
	20'	2-RB-29
	50'	2-RB-30
	50'	2-RB-31
	50'	2-RB-32
	50'	2-RB-33
	50'	2-RB-34
	50'	2-RB-35
	67'	2-RB-48A
	80'	2-RB-36
	80'	2-RB-39
	80'	2-RB-41
	80'	2-RB-43
	80'	2-RB-44
	80'	2-RB-45
	98'	2-RB-37
	117'	2-RB-38
	117'	2-RB-40
	117'	2-RB-42
	117'	2-RB-46
	117'	2-RB-47
	117'	2-RB-48
AOG Building	23'	2-AOG-57
	23'	2-AOG-58
	23'	2-AOG-59
	23'	2-AOG-60
	37'	2-AOG-62
	49'	2-AOG-61
Radwaste Building	-3'	RW-49
	-3'	RW-50
	-3'	RW-51
	23'	RW-52
	23'	RW-53
	23'	RW-54
	23'	RW-55
	23'	RW-56

TABLE 3.7.7.4-1 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Diesel Generator Building	2'	DGB-1
	2'	DGB-2
	2'	DGB-3
	23'	DGB-4
	23'	DGB-5
	23'	DGB-6
	23'	DGB-7
	23'	DGB-8
	23'	DGB-9
	50'	DGB-10
	50'	DGB-11
	50'	DGB-12
	50'	DGB-13
	50'	AFFF HR-2
	50'	AFFF HR-3
Service Water Building	4'	SW-1
	20'	SW-2
	20'	SW-3
Control Building	23'	2-CB-1
	49'	2-CB-2
	70'	2-CB-3
Diesel Generator Tank Area	NA	AFFF HR-1

PLANT SYSTEMS

FOAM SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.5 The following foam systems shall be OPERABLE:

a. Diesel Generator Fuel Oil Tank Area Foam System with:

1. The concentrate proportioning and storage subsystem OPERABLE with 240 gallons of concentrate.

2. Each tank room subsystem OPERABLE.

b. Diesel Generator Air Filter Foam System with:

1. The concentrate proportioning and storage subsystem OPERABLE with 40 gallons of concentrate.

2. Each air filter subsystem OPERABLE.

APPLICABILITY: Whenever the diesel generators are required to be OPERABLE.

ACTION:

a. With one tank room subsystem inoperable, verify the OPERABILITY of the backup foam hose reel within one hour.

b. With one air filter subsystem inoperable, verify the OPERABILITY of two backup foam hose reels within one hour.

c. With any inoperability other than as provided in a and b, above, verify the availability of backup fire suppression equipment for the unprotected area(s) within one hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.7.5 Each of the above required foam systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. A visual inspection of the spray headers to verify their integrity.

3. A visual inspection of each nozzle's spray area to verify that the spray pattern is not obstructed.

4. Conducting a performance evaluation of the concentrate.

PLANT SYSTEMS

3/4.7.8 FIRE BARRIER PENETRATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.8 All fire barrier penetrations, including cable penetration barriers, fire doors, and fire dampers, in fire zone boundaries protecting safety-related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations nonfunctional, within one hour establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the nonfunctional fire barrier and establish an hourly fire watch patrol. Restore the nonfunctional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the nonfunctional penetration, and plans to schedule for restoring the fire barrier penetration(s) to functional status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8 Each of the above required fire barrier penetrations shall be verified to be functional:

a. At least once per 18 months by a visual inspection, and

b. Prior to restoring a fire barrier penetration to functional status following repairs or maintenance, by performance of a visual inspection of the affected fire barrier penetration.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits, per unit, between the offsite transmission network and the onsite Class 1E distribution system, and

b. Four separate and independent diesel generators, each with:

   1. A separate engine-mounted fuel tank containing a minimum of 100 gallons of fuel,

   2. A separate day fuel tank containing a minimum of 22,650 gallons of fuel, and

   3. A separate fuel transfer pump.

c. A plant fuel storage tank containing a minimum of 74,000 gallons of fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION

a. With one offsite circuit of the above required A.C. electrical power** sources not capable of supplying the Class IE distribution system:

   1. Demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.1.a within 2 hours and at least once per 12 hours thereafter;

   2. Demonstrate the OPERABILITY of the diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours and at least once per 72 hours thereafter;

   3. Restore the inoperable offsite circuit to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

** With Unit 1 in OPERATIONAL CONDITION 4 or 5 and one of the required Unit 1 offsite power circuits not capable of supplying the Unit 1 Class 1E distribution system, either restore the inoperable Unit 1 offsite circuit to OPERABLE status within 45 days or place Unit 2 in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of ACTIONS 3.8.1.1.a.1, 3.8.1.1.a.2, and 3.8.1.1.a.3 are not applicable.

ACTION (Continued):

b. With a diesel generator of the above required A.C. electrical power source inoperable*:

* A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

1. Demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 2 hours and at least once per 12 hours thereafter;

2. Demonstrate the OPERABILITY of the remaining diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours and at least once per 72 hours thereafter;

3. Restore the inoperable diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable:

1. Demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a, 4.8.1.1.2.a.4, and 4.8.1.1.2.a.5 within 2 hours and at least once per 12 hours thereafter;

2. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours;

3. With the inoperable offsite A.C. power source restored, demonstrate the OPERABILITY of the remaining A.C. power sources as required by ACTION b; restore four diesel generators to OPERABLE status within 7 days from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours;

4. With the inoperable diesel generator restored, demonstrate the OPERABILITY of the remaining A.C. power sources as required by ACTION a; restore two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With two of the above required offsite A.C. circuits inoperable:

1. Demonstrate the OPERABILITY of four diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within two hours and at least once per 12 hours thereafter, unless the diesel generators are already operating;

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. Restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours;

3. With one offsite source restored, demonstrate the OPERABILITY of the remaining A.C. power sources as required by ACTION a; restore two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With two of the above required diesel generators inoperable:

1. Demonstrate the OPERABILITY of the remaining A.C. power sources by performing Surveillance Requirements 4.8.1.1.1.a, 4.8.1.1.2.a.4, and 4.8.1.1.2.a.5 within 2 hours and at least once per 12 hours thereafter;

2. Restore at least three diesel generators to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours;

3. With one diesel generator restored, demonstrate the OPERABILITY of the remaining A.C. power sources as required by ACTION b; restore at least 4 diesel generators to OPERABLE status within 7 days from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring unit power supply from the normal circuit to the alternate circuit.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the engine-mounted fuel tank,

2. Verifying the fuel level in the day fuel tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the day tank to the engine mounted tank,

4. Verifying the diesel starts and accelerates to at least 514 rpm in less than or equal to 10 seconds,*

5. Verifying the generator is synchronized, loaded to greater than or equal to 1750 kw, and operates for greater than or equal to 15 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

b. At least once per 31 days by verifying the fuel level in the plant fuel storage tank.

c. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM-D975-74 when checked for viscosity, water and sediment,

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,**

2. Verifying the generator capability to reject a load equal to one core spray pump without tripping,

---

* The diesel generator start (10 seconds) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by a manually initiated engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

** For Cycle 9 only, the surveillance interval for Technical Specification 4.8.1.1.2.d.1 may be extended until November 21, 1991.

SURVEILLANCE REQUIREMENTS (Continued)

3. Simulating a loss of offsite power in conjunction with an emergency core cooling system test signal, and:

   a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

   b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected loads through the load sequence relays and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads.

4. Verifying that on the emergency core cooling system test signal, all diesel generator trips except engine overspeed, generator differential, low lube oil pressure, reverse power, loss of field and phase overcurrent with voltage restraint, are automatically bypassed.

5. Verifying the diesel generator operates for greater than or equal to 60 minutes while loaded to greater than or equal to 3500 kw.

6. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3850 kw.

7. Verifying that the automatic load sequence relays are OPERABLE with each load sequence time within 10% of the required value.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit per Unit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two diesel generators, one of which shall be diesel generator 3 or 4, as required to operate ECCS systems in accordance with Specifications 3.5.3.1 and 3.5.3.2:

   1. Each with a separate:

      a) Engine-mounted fuel tank containing a minimum of 100 gallons of fuel,

      b) Day fuel tank containing a minimum of 22,650 gallons of fuel, and

      c) Fuel transfer pump.

   2. With a fuel storage tank containing a minimum of 37,000 gallons of fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving irradiated fuel handling, CORE ALTERATIONS, positive reactivity changes, or operations that have the potential of draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE per surveillance requirements of Specifications 4.8.1.1.1 and 4.8.1.1.2, except for the requirement of 4.8.1.1.2.a.5.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATION OF ONE OR BOTH UNITS

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical buses shall be OPERABLE with tie breakers open between redundant buses:

4160-volt Emergency Bus # E1 and E3

4160-volt Emergency Bus # E2 and E4

480-volt Emergency Bus # E5 and E7

480-volt Emergency Bus # E6 and E8

120-volt A.C. Vital Bus # 1E5 and 2E7

120-volt A.C. Vital Bus # 1E6 and 2E8

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

With less than the above complement of A.C. buses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. buses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

A.C. DISTRIBUTION - SHUTDOWN OF BOTH UNITS

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical buses shall be OPERABLE for each unit but aligned to an OPERABLE diesel generator, as required to operate ECCS systems in accordance with Specifications 3.5.3.1 and 3.5.3.2:

1 - 4160-volt Emergency Bus,

1 - 480-volt Emergency Bus,

2 - 120-volt A.C. Vital Buses.

APPLICABILITY: CONDITIONS 4 and 5.

ACTION:

With less than the above complement of A.C. buses OPERABLE, suspend all operations involving irradiated fuel handling, CORE ALTERATIONS, positive reactivity changes, or operations that have the potential of draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. buses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 As a minimum, the following D.C. divisions shall be OPERABLE with tie breakers between divisions open:

a. Division I, consisting of:
   1. A 250/125 volt bus.
   2. Two 125 volt D.C. batteries, 2A-1 and 2A-2, each with a full capacity charger.

b. Division II, consisting of:
   1. A 250/125 volt bus.
   2. Two 125 volt D.C. batteries, 2B-1 and 2B-2, each with a full capacity charger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more batteries and/or its associated charger inoperable in one division, restore the division to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more batteries and/or its associated charger inoperable in both divisions, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each of the above required D.C. divisions shall be determined OPERABLE with tie breakers open at least once per 7 days by verifying:

a. Correct breaker alignment and indicated power availability, and

b. That no combination of more than two power conversion modules, consisting of either two lighting inverters or one lighting inverter and one plant UPS unit, are aligned to division II bus B.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.2.3.2 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8.2.3.2-1 meet the Category A limits, and

2. Total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At least once per 92 days by verifying that:

1. The parameter in Table 4.8.2.3.2-1 meet the Category B limits

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of the connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material, and

3. The battery charger will supply at least 250 amperes at a minimum of 135 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that:

1. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to 105 volts.

a) During the initial 60 seconds of the test:

1) Battery 2A-1 greater than or equal to 916 amperes.
2) Battery 2A-2 greater than or equal to 916 amperes.
3) Battery 2B-1 greater than or equal to 916 amperes.
4) Battery 2B-2 greater than or equal to 916 amperes.

b) During the remainder of the first 30 minutes of the test:

1) Battery 2A-1 greater than or equal to 250 amperes.
2) Battery 2A-2 greater than or equal to 250 amperes.
3) Battery 2B-1 greater than or equal to 250 amperes.
4) Battery 2B-2 greater than or equal to 250 amperes.

c) During the remainder of the 4 hour test:

1) Battery 2A-1 greater than or equal to 200 amperes.
2) Battery 2A-2 greater than or equal to 200 amperes.
3) Battery 2B-1 greater than or equal to 200 amperes.
4) Battery 2B-2 greater than or equal to 200 amperes.

2. At the completion of the above tests, the battery charger shall be demonstrated capable of recharging its battery at a rate of at least 200 amperes while supplying normal D.C. loads. The battery shall be charged to at least 95% capacity in less than or equal to 24 hours.

e. At least once per 60 months during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test.

TABLE 4.8.2.3.2-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[(1)]	CATEGORY B[(2)]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[(3)] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ 1/4" above maximum level indication mark	>Minimum level indication mark, and $\leq$ 1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts[(c)]	$\geq$ 2.07 volts
		any one cell $\geq$ 1.195	Not more than .020 below the average of all connected cells
Specific Gravity[(a)]	$\geq$ 1.200[(b)]	Average of all connected cells $\geq$ 1.205	Average of all connected cells $\geq$ 1.195[(b)]

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps.

(c) Corrected for average electrolyte temperature.

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 7 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

D.C. DISTRIBUTION - OPERATION OF ONE OR BOTH UNITS

LIMITING CONDITION FOR OPERATION

3.8.2.4.1 The 125 VDC control power circuits shall be OPERABLE from their normal source for the following equipment:

a. Diesel Generator #1, 4160 V emergency bus E1, and 480 V emergency bus E5.

b. Diesel Generator #2, 4160 V emergency bus E2, and 480 V emergency bus E6.

c. Diesel Generator #3, 4160 V emergency bus E3, and 480 V emergency bus E7.

d. Diesel Generator #4, 4160 V emergency bus E4, and 480 V emergency bus E8.

e. ESS panel H58*

f. ESS panel H59*

g. ESS panel H60*

h. ESS panel H61*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With 125 VDC control power circuit for the diesel generator, 4160 V emergency bus, or 480 V emergency bus not OPERABLE from its normal source, declare the affected diesel generator, 4160 V emergency bus, or 480 V emergency bus inoperable and either:

   1. Take the applicable ACTION statement for the inoperable equipment, or

   2. Declare the affected equipment OPERABLE by manually transferring the 125 VDC control power circuit for the affected diesel generator, 4160 V emergency bus, or 480 V emergency bus to the OPERABLE alternate source.

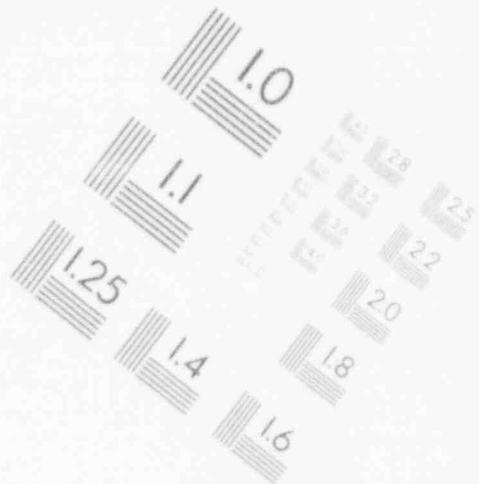
b. With the 125 VDC control power circuit for ESS panels H58, H59, H60, or H61 not OPERABLE from its normal source, either:

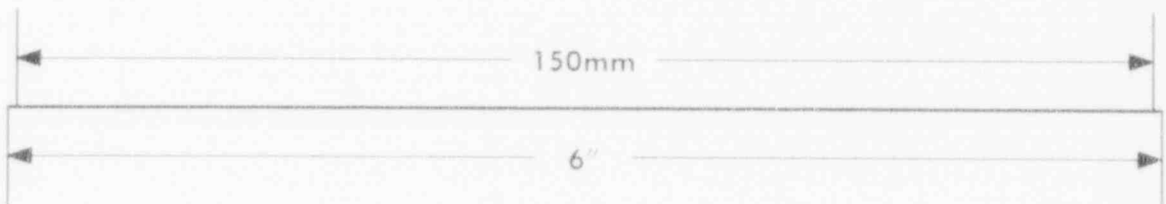
   1. Verify the alternate source is OPERABLE and that power availability is indicated, or

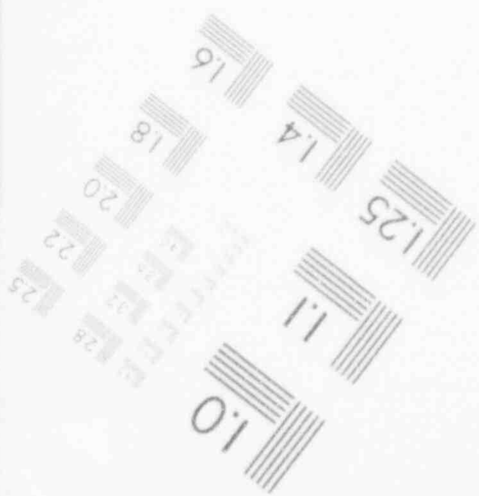
* The ESS panel automatically transfers to its alternate source should the normal source de-energize. Refer to ACTION b.

2

IMAGE EVALUATION TEST TARGET (MT-3)







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770 BASKET ROAD
P.O. BOX 3?8
WEBSTER, NEW YO[illegible] 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

PHOTOGRAPHIC SCIENCES CORPORATION
770 BASKET ROAD
P.O. BOX 338
WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

PHOTOGRAPHIC SCIENCES CORPORATION
770 BASKET ROAD
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(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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(716) 265-1600

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. When the alternate source is inoperable, declare the affected equipment inoperable and take the applicable ACTION statement for the inoperable equipment, or

3. Verify the 125 VDC control power circuit for the affected ESS panel has automatically transferred to its OPERABLE alternate source and that power availability is indicated.

c. Restore the affected 125 VDC control power circuit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1.1 The above specified normal 125 volt D.C. control power circuits shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.1.2 The batteries and chargers associated with the above normal 125 volt D.C. control power circuits shall be determined OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4.2 As a minimum, Division I or Division II of the D.C. power distribution system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than Division I or Division II of the above required D.C. distribution system OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, and all operations that could decrease SHUTDOWN MARGIN or have the potential for draining the reactor vessel. Restore at least one division to OPERABLE status within 7 days.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.4.2.1 The above required D.C. Division shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2.2 The batteries and chargers associated with the above required division shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

*When handling irradiated fuel in the secondary containment.

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.2.5 Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE.

APPLICABILITY: Whenever the respective power supply is supplying power to a RPS bus.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.2.5 The above specified RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST, and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage, and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic, and output circuit breakers and verifying the following setpoints:

	RPS MG SET	ALTERNATE SOURCE
1. Over-voltage	$\leq$ 129 VAC	$\leq$ 132 VAC
2. Under-voltage	$\geq$ 105 VAC	$\geq$ 108 VAC
3. Under-frequency	$\geq$ 57 Hz	$\geq$ 57 Hz

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Refuel position* with at least the Refuel position "one-rod-out" interlock OPERABLE and with the following Refuel position equipment interlocks OPERABLE when equipment associated with the interlock is being operated for CORE ALTERATIONS:

a. All rods in.
b. Refuel platform position.
c. Refuel platform hoists fuel-loaded.
d. Fuel grapple position.
e. Service platform hoist fuel-loaded.

APPLICABILITY: CONDITION 5.

ACTION:

a. With the reactor mode switch not locked in the Refuel position or the one-rod-out interlock inoperable, immediately suspend all CORE ALTERATIONS.

b. With any of the above required Refuel position equipment interlocks inoperable, suspend operations with equipment associated with the inoperable Refuel position equipment interlock.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Refuel position:

a. Within 2 hours prior to beginning CORE ALTERATIONS,

b. Prior to resuming CORE ALTERATIONS when the reactor mode switch has been unlocked, and

c. At least once per 12 hours.

*See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks shall be demonstrated OPERABLE:

a. Within 24 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS, and

b. Prior to resuming CORE ALTERATIONS following repair, maintenance, or replacement of any component that could affect the Refuel position interlocks.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 During CORE ALTERATIONS, the requirements for the source range monitors (SRMs) shall be:

a. Two SRMs* shall be OPERABLE, one in the core quadrant where fuel is being moved and one in an adjacent quadrant. For an SRM to be considered OPERABLE, it shall be inserted to the normal operating level and shall have a minimum of 3 cps except as specified in d and e below.

b. The SRMs shall give a continuous visual indication in the Control Room.

c. The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn** and shutdown margin demonstrations.

d. During a core SPIRAL UNLOAD the count rate may drop below 3 cps.

e. Prior to a core SPIRAL RELOAD, up to four fuel assemblies shall be loaded into different control cells containing control blades around each SRM to obtain 3 cps. Until these assemblies have been loaded, the 3 cps count rate is not required.

APPLICABILITY: OPERATIONAL CONDITION 5

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and fully insert all insertable control rods. The provisions of Specification 3.0.3 are not applicable.

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* The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

** Not required for control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours;

   1. Performance of a CHANNEL CHECK,

   2. Verifying the detectors are inserted to the normal operating level,

   3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and one is located in the adjacent quadrant,

   4. During CORE ALTERATIONS, verifying that the channel count rate is at least 3 cps (except as noted in Specification 3.9.2.d and 3.9.2.e),

   5. During a core SPIRAL UNLOAD or SPIRAL RELOAD, verifying that the fuel movement sheet is being followed.

b. Verifying prior to the start of a SPIRAL RELOAD that the SRMs have been raised to a count rate of at least 3 cps by the insertion of up to four fuel assemblies around each of the four SRMs.

c. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to the start of CORE ALTERATIONS, and

   2. At least once per seven days.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be fully inserted*.

APPLICABILITY: CONDITION 5, during CORE ALTERATIONS**.

ACTION:

With all control rods not fully inserted, immediately deenergize the control rod scram solenoid valves. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 Verify all control rods to be fully inserted within 2 hours prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

*Except control rods removed per Specification 3.9.10.1 or 3.9.1.2.

**See Special Test Exception 3.10.3.

3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 24 hours, suspend movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling platform personnel.

APPLICABILITY: CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and the refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 CRANE AND HOIST OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 All cranes and hoists used for handling fuel assemblies or control rods within the reactor pressure vessel shall be OPERABLE.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for crane or hoist OPERABILITY not satisfied, suspend use of any inoperable crane or hoist from operations involving the movement of control rods and fuel assemblies after placing the load in a safe location. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 Each crane or hoist used for movement of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff when the load exceeds 1600 pounds for the mast fuel gripper and $\leq$ 1050 pounds for all other cranes and hoists.

b. Demonstrating operation of the loaded interlock when the load exceeds 750 pounds for the mast fuel gripper and $\leq$ 350 pounds for all other cranes and hoists.

c. Demonstrating operation of the uptravel stop for all cranes and hoists other than the mast fuel gripper when uptravel would bring the top of the active fuel to 7 feet below the normal spent fuel pool water level.

d. Demonstrating operation of the slack cable cutoff when the load is less than 50 $\pm$ 25 pounds for the mast fuel gripper.

e. Performing a load test of at least 1000 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1600 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Loads other than fuel assemblies shall be verified to be $\leq$ 1600 pounds prior to movement over fuel assemblies in the spent fuel storage pool racks.

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 23 feet of water shal be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: CONDITION 5, during movement of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during movement of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 20 feet 6 inches of water shall be maintained over the top of the irradiated fuel rods seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the load in a safe location. Restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth at least once per 7 days.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or its associated control rod drive mechanism may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until the control rod and control rod drive mechanism are reinstalled and the control rod is fully inserted in the core:

a. The reactor mode switch is locked in the Refuel position per Specification 3.9.1,

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2,

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied prior to the removal of the control rod,

d. All other control rods in a five-by-five array centered on the control rod being removed are fully inserted and electrically disarmed, and

e. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or control rod drive mechanism and initiate action to satisfy the above requirements. The provisions of Specification 3.0.3 are not applicable.

RETYPED TECH. SPECS.
Updated Thru. Amend.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or its associated control rod drive mechanism and at least once per 24 hours thereafter until the control rod and control rod drive mechanism are reinstalled and the control rod is fully inserted in the core, verify that:

a. The reactor mode switch is locked in the Refuel position with at least the "one-rod-out" Refuel interlock OPERABLE per Specification 3.9.1,

b. The SRM channels are OPERABLE per Specification 3.9.2,

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied as above specified,

d. All other control rods in a five-by-five array centered on the control rod being removed are fully inserted and electrically disarmed, and

e. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed.

RETYPED TECH. SPECS.
Updated Thru. Amend.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the reactor pressure vessel, provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core:

a. The reactor mode switch is locked in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are fully inserted.

e. The four fuel assemblies are removed from the core cell surrounding each control rod or control rod drive mechanism to be removed.

APPLICABILITY: CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms and initiate action to satisfy the above requirements. The provisions of Specification 3.0.3 are not applicable.

RETYPED TECH. SPECS.
Updated Thru. Amend. 7

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Refuel position per Specification 3.9.1,

b. The SRM channels are OPERABLE per Specification 3.9.2,

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied,

d. All other control rods are fully inserted, and

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism that is to be removed from the reactor vessel at the same time are removed from the core.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

RETYPED TECH. SPECS.
Updated Thru. Amend. 78