U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-325, 50-324
License Nos.:	DPR-71, DPR -62
Report Nos.:	50-325/97-300, 50-324/97-300
Licensee:	Carolina Power and Light Company
Facility:	Brunswick Steam Electric Plant Units 1 & 2
Location:	Southport, NC
Dates:	April 25 through May 2, 1997
Examiners:	George T. Hopper, Chief License Examiner
	James H. Moorman, III, License Examiner
	D. Charles Payne, License Examiner
Approved by:	Thomas A. Peebles, Chief,
	Operator Licensing and Human Performance Branch
	Division of Reactor Safety

Enclosure 1

EXECUTIVE SUMMARY

Brunswick Steam Electric Plant Units 1 & 2
NRC Examination Report Nos. 50-325/97-300, 50-324/97-300

During the period April 25 through May 2, 1997, NRC examiners conducted an announced operator licensing initial examination in accordance with the guidance of Examiner Standards, NUREG-1021, Interim Revision 8. This examination implemented the operator licensing requirements of 10 CFR §55.41, §55.43, and §55.45.

Operations

- Control room activities were observed during the examination validation week and examination administration week. The operators were found to be attentive and professional in their duties (Section O1.1).

- Five Senior Reactor Operator (SRO) candidates and four Reactor Operator (RO) candidates received written examinations and operating tests. The licensee administered the written examination on April 25, 1997, and the NRC administered the operating tests April 28 - May 2, 1997 (Section O5.1).

- Four SRO and three RO candidates passed the examination. Five of the seven candidates marginally passed the examination. One SRO and one RO candidate failed. (Section O5.1)

- Candidate Pass/Fail

	SRO	RO	Total	Percent
Pass	4	3	7	77.8%
Fail	1	1	2	22.2%

- Licensee examination preparation activities were considered good (Section O5.2).

- The examiners concluded that overall candidate performance on the operating test was weak. Candidates had difficulty prioritizing and correctly implementing the mitigation strategies of the emergency operating procedures (Section O5.3).

- The examiners identified a generic performance weakness in the area of procedural compliance (Section O5.3).

- One deviation concerning the description of the operation of the Automatic Depressurization System was identified (Section O8.1).

Report Details

Summary of Plant Status

During the period of the examinations Unit 1 and Unit 2 were at 100 percent power.

I. Operations

O1 Conduct of Operations

O1.1 Control Room Observation

During validation and administration of the examination, the examiners observed the conduct of operations by currently licensed operators in the control room. The ROs were attentive to the evolutions in progress. The SROs limited personnel access for official business only, which contributed to a quiet, professional atmosphere.

O5 Operator Training and Qualifications

O5.1 General Comments

NRC examiners conducted regular, announced operator licensing initial examinations during the period April 25 - May 2, 1997. NRC examiners administered examinations developed by the licensee's training department, under the requirements of an NRC security agreement, in accordance with the guidelines of the Examiner Standards (ES), NUREG-1021, Interim Revision 8. Three SRO instant, two SRO upgrade, and four RO license applicants received written examinations and operating tests.

Four SRO and three RO candidates passed the examination. Five of the seven candidates marginally passed the examination. One SRO and one RO candidate failed. Two candidates (one SRO and one RO) were graded as marginal passes on the Job Performance Measure (JPM) portion of the operating test. One SRO candidate was a marginal pass on both the administrative portion and the dynamic simulator portion of the operating test. One RO candidate marginally passed the dynamic simulator. Yet a fourth SRO candidate was graded as a marginal pass on all three categories of the operating test. Candidate's are considered to have marginally passed if they receive an unsatisfactory grade on any one administrative topic area, complete only 80 percent of the JPMs successfully , or receive a grade of 1.8 to 2.0 on any one competency during the dynamic simulator examinations. Detailed candidate performance comments have been transmitted under separate cover for management review and to allow appropriate candidate remediation.

O5.2 Pre-Examination Activities

The licensee developed the SRO and RO written examinations, three JPM sets, and four dynamic simulator scenarios for use during this examination. All materials were submitted to the NRC on time and were of good quality, meeting the guidelines specified in NUREG-1021. Most of the changes made to the written examinations were editorial in nature. Only two questions out of a total of 125 contained distractors which needed to be altered and one additional question contained two potentially

correct answers. The NRC reviewed the written examinations and found that 76 percent of the questions were written at the comprehension/analysis level. A total of 62 new questions had been developed.

The simulator scenarios were challenging and designed to ensure that each candidate could be adequately evaluated on a majority of the items listed in 10 CFR 55.45(a). The examiners considered each scenario to be a challenging test of the candidates' ability. Few changes were made to the content of each scenario. However, the NRC requested additional detail be provided in the operator activities section of some of the scenarios to clarify expected operator actions.

O5.3 Examination Results and Related Findings, Observations, and Conclusions

a. Scope

The examiners evaluated the candidates' compliance with and use of plant procedures during the simulator scenarios and JPMs. The guidelines of NUREG-1021, Forms ES-303-3 and ES-303-4, "Competency Grading Worksheets for Integrated Plant Operations," were used as a basis for the evaluations.

b. Observations and Findings.

Examiners identified numerous weaknesses in candidate performance during the operations portion of the examination. Details of the discrepancies are described in each individual's examination report, Form ES-303-1, "Operator Licensing Examination Report". The examiners identified several generic weaknesses which were of particular concern.

During performance of one scenario, the candidates responded to a recirculation line break Loss Of Coolant Accident (LOCA). Two out of three crews failed to initiate an emergency depressurization when required upon reaching Top of Active Fuel (TAF). The crews did not manually open Automatic Depressurization System (ADS) valves until level had decreased below the Minimum Zero Injection Level (LL5). The scenario involved a loss of reactor coolant which exceeded the injection flowrate of the High Pressure Coolant Injection (HPCI) pump. Level decreased to TAF in approximately two minutes and ADS valves were not opened until four minutes after the start of the event. The candidates should have anticipated the need for an emergency depressurization, based upon the rate of level decrease, and opened the ADS valves when level reached TAF per step RC/L-28 of Emergency Operating Procedure EOP-01-RVCP. Emergency depressurization should have been manually initiated approximately two minutes sooner than it actually occurred. Examiners observed one crew conduct a crew brief as level continued to decrease before the directive was given to open the ADS valves. The order to open the valves should have been given immediately. While both crews were observed to have been aggressively performing the actions of EOP-02-PCCP, Primary Containment Control Procedure", addressing adverse containment conditions, the priority of ensuring adequate core cooling was neglected.

During the performance of emergency depressurization, two of the three crews failed to ensure that Core Spray Loop A was injecting into the Reactor Pressure Vessel (RPV) as required by Procedure EOP-01-RVCP. Only two sources of Emergency Core Cooling System (ECCS) low pressure injection were available during the event. Low Pressure Coolant Injection (LPCI) Loop A, had been lined up earlier for suppression chamber spray, but was available for injection. Core Spray Loop A, was also available but had to be manually started and lined up due to a LOCA logic failure. Both crews did not open the core spray injection valve (F-005A) until after the RPV had been depressurized. One crew took more than five minutes to realize that core spray was not injecting.

Adequate core cooling exists so long as RPV water level remains above TAF. If at least one source of injection into the RPV is available, emergency depressurization is required when water level reaches TAF to maximize the injection flowrate from all operating sources of injection. The consequences of not depressurizing the RPV under conditions that require an emergency depressurization could include a loss of adequate core cooling or failure of the primary containment. Procedure EOP-01-RVCP required at least two low pressure injection systems be lined up to obtain maximum available flow to provide core cooling and restore vessel level above TAF. The examiners noted that the most significant operator error identified in the BSEP Probabilistic Safety Assessment (PSA) is when operators fail to manually depressurize the reactor when required. Therefore, this is of concern because an increase in likelihood that operators fail to depressurize prior to reaching TAF would have a large effect on the PSA results and the plants's calculated Core Damage Frequency (CDF).

The examiners observed other misapplications of procedures or oversights by the candidates. None of the crews individually scramed a drifting rod as instructed by Procedure APP-A-05 3-2, "ROD DRIFT" step 4.e. Two crews failed to follow a procedural precaution and secure the RCIC pump to prevent continuous operation at low speed despite this precaution being posted as an operator and on the control panel. One crew failed to inhibit ADS when reactor water level decreased below +45 inches as required by Procedure EOP-01 step RC/L-22, and allowed the reactor to automatically depressurize. This unplanned emergency depressurization may have altered the mitigation strategy of the event had the operators had time to recover HPCI and regain control of RPV level.

Candidate performance on the walkthrough portion of the examination was poor. Four out of nine candidates marginally passed this portion of the examination. One other candidate failed. The examiners noted performance errors ranging from inattention to detail to gross conceptual errors. Examples of some of these errors included:

- Racking in the wrong breakers to cross-tie emergency busses.

- Using a number greater than 100 percent in an attempt to calculate alternate power indication per Procedure GP-3 using bypass valve position.

- Demonstrated unfamiliarity with Diesel Generator controls and proper operation of the diesel.

- Improperly venting the drywell instead of the suppression chamber given the plant conditions imposed.
- Improperly performing procedure LEP-02, "Alternate Control Rod Insertion." Candidates scrammed rods simultaneously vice individually.
- Performing Alternate Emergency Depressurization procedure too slowly with little concern for vessel level or the need for adequate core cooling.

The examiners noted that candidates were unfamiliar with some of the procedures and encountered difficulty in ensuring verbatim compliance. Most of the discrepancies noted could be attributed to the candidates failure to carefully read and follow procedural guidance.

c. Conclusion

The examiners concluded that overall candidate performance on the operating test was weak. Candidates had difficulty prioritizing and correctly implementing the mitigation strategies of the EOPs and were in sensitive to the need to keep the core covered in order to ensure adequate core cooling. Lack of procedural compliance contributed to many of the errors that were observed. The weaknesses noted were fundamental in nature and should not be attributed to subtleties or the level of difficulty of any part of the operating test.

O8 Miscellaneous Operations Issues

O8.1 Special Review of UFSAR Commitments

a. Scope

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the examinations discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas that were examined. The inspectors intent was to verify that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

b. Observations and Findings

The inspector found that the system description for the Automatic Depressurization System on page 1.2.2-9 indicated that the system will function automatically in a LOCA situation in which the HPCI system fails to automatically maintain reactor vessel water level. Current procedural guidance contained in EOP-01-RVCP (step RC/L-22) has the operators inhibiting ADS when vessel level drops to +45 inches, thereby defeating the automatic operation of the system. Consequently, the system will not operate as stated unless the operators do not perform the actions required by the EOPs. In addition, FSAR Chapter 15 page 15.OA.6-15, Event 34-Pipe Breaks Inside Primary Containment," described the equipment required to mitigate a LOCA which

included the ADS system. This section again refers to the automatic operation of ADS and does not indicate that the system's automatic feature will be defeated and operated manually by the operators during the blowdown phase of the event.

c. Conclusion

The inspectors concluded that the UFSAR wording in the items mentioned above were inconsistent with current plant operating practices and procedures . These discrepancies are collectively identified as additional examples of URI 50-325,324/96-05-02, "FSAR Discrepancies."

V. Management Meetings

X1. Exit Meeting Summary

At the conclusion of the site visit, the examiners met with representatives of the plant staff listed on the following page to discuss the results of the examinations.

The examiners asked the licensee whether any materials examined should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

L. Dunlap, Training Manager (Acting)
J. Gawron, Manager NAD
K. Jury, Manager Regulatory Affairs
W. Levis, Director of Site Operations
R. Lopriore, Plant Manager
K. McCall, Supervisor, Operator Initial Training
R. Mullis, Operations Manager
J. Rainsburrow, Operations Support Manager
G. Thearling, Regulatory Affairs

NRC

E. Brown, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

NONE

Discussed

96-05-02 FSAR Discrepancies

NONE

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
CFR	Code of Federal Regulations
ECCS	Emergency Core Cooling Systems
ES	Examiner Standards (NUREG-1021)
HPCI	High Pressure Coolant Injection
JPM	Job Performance Measure
LPCI	Low Pressure Coolant Injection
NRC	Nuclear Regulatory Commission
RCIC	Reactor Core Isolation Cooling
RO	Reactor Operator
RPV	Reactor Pressure Vessel
SRO	Senior Reactor Operator
TAF	Top of Active Fuel
UFSAR	Updated Final Safety Analysis Report

SIMULATION FACILITY REPORT

Facility Licensee: Brunswick Steam Electric Plant

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

Operating Tests Administered on: April 28 - May 2,

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

While conducting the simulator portion of the operating tests, the following items were observed (if none, so state):

ITEM DESCRIPTION

NONE

SIMULATION FACILITY REPORT

Facility Licensee: Brunswick Steam Electric Plant

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

Operating Tests Administered on: April 28 - May 2,

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

While conducting the simulator portion of the operating tests, the following items were observed (if none, so state):

<u>ITEM</u> <u>DESCRIPTION</u>

NONE

WRITTEN EXAMINATION(S) AND ANSWER KEY(S) (SRO/RO)

Enclosure 3

NRC
Master

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information	
Name:	Region: II
Date: 04/25/97	Facility/Unit: Brunswick/1 & 2
License Level: Senior Reactor Operator	Reactor Type: GE BWR-4
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected four hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results	
Examination Value	_________ Points
Applicant's Score	_________ Points
Applicant's Grade	_________ Percent

WRITTEN EXAMINATION GUIDELINES

1. 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.

2. To pass the examination, you must achieve a grade of 80.00 percent or greater. Every question is worth one point.

3. For an initial examination, the time limit for completing the examination is four hours.

4. You may bring pens and calculators into the examination room. Use only black ink to ensure legible copies.

5. Print your name in the blank provided on the examination cover sheet and the answer sheet. You may be asked to provide the examiner with some form of positive identification.

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If the intent of a question is unclear, ask questions of the NRC examiner or the designated facility instructor only.

8. 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.

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

10. After you have turned in your examination, leave the examination area as defined by the proctor or NRC examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

11. Do you have any questions?

QUESTION 1 POINT VALUE: 1.00

Unit One (1) is operating with the following plant conditions:

Core Thermal Power	2558 MWth
Reactor pressure	1030 psig
Core Flow	77 Mlbm/hr

The SAFETY LIMIT for THERMAL POWER is the MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than:

a. 1.07

b. 1.08

c. 1.09

d. 1.10

QUESTION 2 POINT VALUE: 1.00

A Unit Two (2) startup is in progress per GP-02. The Reactor is critical with Reactor power at the point of adding heat.

Coolant temperature is being raised to saturation conditions, with Reactor steam dome pressure at 0 psig. RWCU is in service.

2PT-01.7, Heatup/Cooldown Monitoring is being performed. Which of the following conditions would result in **UNSATISFACTORY** Acceptance Criteria of the PT?

a. G31-TI-R607, Pt 5 indicates 169°F
   C12-TR-R018, Ch 151 indicates 164°F
   Reactor water level is 200"

b. G31-TI-R607, Pt 5 indicates 187°F
   C12-TR-R018, Ch 151 indicates 183°F
   Reactor water level is 210"

c. G31-TI-R607, Pt 5 indicates 174°F
   C12-TR-R018, Ch 151 indicates 170°F
   Reactor water level is 200"

d. G31-TI-R607, Pt 5 indicates 198°F
   C12-TR-R018, Ch 151 indicates 195°F
   Reactor water level is 210"

QUESTION 3 POINT VALUE: 1.00

Unit One (1) is operating with the following conditions:

Reactor Power	55%
Reactor Feed Pump 1A	operating
Reactor Feed Pump 1B	idling
Recirculation Pump Speeds	58%

Reactor Feed Pump (RFP) 1A trips and Reactor Level Hi/Lo alarms. Reactor Level drops to the scram setpoint and continues to lower to +110 inches before the operator brings the idling RFP on line (30 seconds after the trip of RFP 1A) to restore Reactor Level.

What should be the present status of the Recirculation Pumps?

a. Running at 58% speed.

b. Running on limiter #2.

c. Running on limiter #1.

d. Tripped on ATWS ARI/RPT.

QUESTION 4 POINT VALUE: 1.00

Following unsatisfactory performance of PT-13.1, the crew has entered AOP-04.4. One jet pump has been declared INOPERABLE, requiring a plant shutdown per GP-05 and Technical Specifications.

Technical Specification Bases requires a plant shutdown due to the hazard in case of a Design Basis Accident associated with the increased blowdown area and the:

a. reduction in core cooling from coastdown flow with a broken jet pump riser.

b. reduction in core cooling from coastdown flow with a broken jet pump standpipe.

c. elimination of the capability of reflooding the core with a broken jet pump riser.

d. elimination of the capability of reflooding the core with a broken jet pump standpipe.

QUESTION 5 POINT VALUE: 1.00

Unit One (1) is operating at 27% power during Unit startup. The Turbine Generator has been synchronized to the grid. A total loss of Division II DC Switchboard 1B results in a reactor scram. During this transient:

57 control rods fail to fully insert
Reactor pressure peaks at 1132 psig
Reactor water level lowers to +107 inches
BOP Buses fail to transfer to the SAT
Diesel Generator 2 fails to start

What is the expected status of the Alternate Rod Injection (ARI) system? ARI has:

a. auto initiated on high reactor pressure.

b. auto initiated on low reactor water level.

c. not auto initiated but can be manually initiated.

d. not auto initiated and cannot be manually initiated.

QUESTION 6 POINT VALUE: 1.00

Following a line break on Unit Two (2) plant conditions are:

Reactor pressure	450 psig
Reactor water level	+70 inches
Drywell pressure	4.8 psig
Drywell temp (average)	165 deg F

All Drywell Cooler Fans will:

a. trip but can be restarted per SEP-10.

b. trip and cannot be restarted per SEP-10.

c. auto start but can be tripped at the RTGB.

d. auto start and cannot be tripped at the RTGB.

QUESTION 7 POINT VALUE: 1.00

Unit Two (2) is in OPERATIONAL CONDITION 5, with CORE ALTERATIONS in progress and SECONDARY CONTAINMENT INTEGRITY established.

The Interruptible Instrument Air Header ruptures and the Interruptible Air Header isolation valves are closed. Non-Interruptible Air Header pressure is normal.

How is SECONDARY CONTAINMENT INTEGRITY affected?

a. Reactor Building Supply and Exhaust Fans remain in service, SBGT auto starts.

b. Reactor Building Supply and Exhaust Fans remain in service, SBGT remains in standby.

c. Reactor Building Supply and Exhaust Fans trip and SBGT auto starts to maintain negative pressure.

d. Reactor Building Supply and Exhaust Fans trip but SBGT must be manually started to maintain negative pressure.

QUESTION 8 POINT VALUE: 1.00

Unit Two (2) is operating at power with Diesel Geneator 3 (DG3) under clearance. A lockout of a BOP Bus initiates a transient resulting in a reactor scram signal and an ATWS. Plant conditions:

Reactor power	10%
Bus E3	De-energized
SLC Switch	PUMP B RUN position
Bus E7-E8	Cross-tie Breakers racked in

How will the SLC system respond when the Bus E7-E8 Cross-tie Breakers are closed? SLC squib valve:

a. A fires, no SLC pump starts.

b. B fires, no SLC pump starts.

c. A fires, one SLC pump starts.

d. B fires, one SLC pump starts.

QUESTION 9 POINT VALUE: 1.00

A Unit One (1) reactor scram occurs from 100% power. The operator completes the immediate scram actions and notes the following Rod Worth Minimizer (RWM) display:

ALL RODS IN:	NO
SHUTDOWN:	YES
RODS NOT FULL IN:	001

When the operator depresses the List Rods RWM key, rod 02-51 is displayed. What can be the furthest withdrawn position of rod 02-51 to cause the above display, and the basis for that rod position?

a. 48, shutdown margin calculations.

b. 06, maximum subcritical bank withdrawal position.

c. 04, maximum subcritical bank withdrawal position.

d. 02, maximum subcritical bank withdrawal position.

QUESTION 10 POINT VALUE: 1.00

During a low water level condition, CRD Flow maximization is being implemented per SEP-09 with the Reactor Building accessible.

The operator is directed to maintain Charging Water Header pressure ≥950 psig while opening the Flow Control and Pressure Control valves.

This limitation will prevent pump:

a. trip on overcurrent protection.

b. trip due to low suction pressure.

c. flowrate in excess of runout capacity.

d. discharge pressure dropping below reactor pressure.

QUESTION 11 POINT VALUE: 1.00

Unit One (1) is in OPERATIONAL CONDITION 5, performing CORE ALTERATIONS. A fuel shuffle is being performed (the core will not be unloaded). The following SRM indications are observed:

SRM Channel A	4 cps
SRM Channel B	6 cps
SRM Channel C	3 cps
SRM Channel D	2 cps

Per Technical Specifications, CORE ALTERATIONS may be performed in the following core quadrants:

a. Northeast and Northwest only.

b. Northeast and Southeast only.

c. Northeast, Northwest and Southeast only.

d. Northeast, Northwest and Southwest only.

QUESTION 12 POINT VALUE: 1.00

During a Reactor startup following refueling, the operator is performing PT-50.2, IRM Range 6/7 Overlap Determination. The following data is recorded:

	Range 6	Range 7
IRM A	50	4
IRM B	42	5
IRM C	45	5
IRM D	43	4
IRM E	37	3
IRM F	52	6
IRM G	44	4
IRM H	50	6

The Level 2 acceptance criteria of PT-50.2 is:

a. satisfactory for all Division I and II IRMs.

b. unsatisfactory for at least one Division I IRM only.

c. unsatisfactory for at least one Divivision II IRM only.

d. unsatisfactory for at least one Division I and one Division II IRM.

QUESTION 13 POINT VALUE: 1.00

APRM Channel C has all associated LPRM inputs Operable, with the LPRM function switches in Operate. During performance of a MST, the LPRM function switches are placed to and left in Bypass, one at a time.

What is the MINIMUM number of LPRM function switches that should have been placed in Bypass when a Neutron Monitoring trip signal (1/2 scram) is received?

a. 3

b. 4

c. 6

d. 7

QUESTION 14 POINT VALUE: 1.00

Unit Two (2) is operating at 100% power. The Standby Gas Treatment System (SBGT) is in the standby alignment. A Trip of both Reactor Feedwater Pumps results in the following plant conditions:

Reactor water level	+80 inches
Reactor pressure	945 psig
Drywell pressure	+0.3 psig
Rx Bldg pressure	-0.4" WC

HPCI and RCIC have initiated to restore Reactor water level. All systems respond as designed during the transient. What operator action is <u>required</u> concerning the SBGT System?

a. Secure one SBGT fan per OP-10.

b. Open Post LOCA Vent valves (SGT-V8 and V9).

c. Open Primary Containment Suction valve (VA-2F-BFV-RB).

d. Restart Rx Bldg HVAC and secure both SBGT trains per SEP-04.

QUESTION 15 POINT VALUE: 1.00

A condition has arisen on Unit Two (2) requiring <u>BOTH</u> of the Reactor Building Vent Exhaust Radiation Monitors to be declared INOPERABLE.

The Shift Superintendent has determined that Unit Two (2) must be placed in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

What additional action is <u>**REQUIRED**</u> by Technical Specifications?

a. Establish SECONDARY CONTAINMENT INTEGRITY with the Standby Gas System operating within one hour.

b. Establish SECONDARY CONTAINMENT INTEGRITY with the Standby Gas System operating within 24 hours.

c. Place at least one INOPERABLE Reactor Building Vent Radiation channel in the tripped condition within one hour.

d. Place at least one INOPERABLE Reactor Building Vent Radiation channel in the tripped condition within 24 hours.

QUESTION 16 POINT VALUE: 1.00

Unit One (1) is operating at rated power, when the loss of an electrical power distribution system results in the following Group 1 PCIS status light indications on P601:

Inboard MSIV DC solenoid	Out
Inboard MSIV AC solenoid	Out
Outboard MSIV DC solenoid	Out
Outboard MSIV AC solenoid	Lit

What power distribution system has been lost?

a. Division I AC.

b. Division I DC.

c. Division II AC.

d. Division II DC.

QUESTION 17 POINT VALUE: 1.00

Unit One (1) is performing Alternate Emergency Depressurization. RCIC has been placed into pressure control to aid in Reactor pressure reduction.

Circuit alterations have been performed per EOP-01-RVCP and EOP-01-SEP-10 for HPCI and RCIC.

Which of the following conditions, by interlock, will remove the RCIC System from the pressure control mode of operation?

a. Reactor pressure lowers to 50 psig.

b. Drywell pressure rises to 2.5 psig.

c. Reactor water level lowers to +110 inches.

d. Suppression pool level rises to -23 inches.

QUESTION 18 POINT VALUE: 1.00

Unit One (1) failed to scram. RHR is placed in Suppression Pool Cooling per the Hard Card without use of overrides. Reactor water level is then deliberately lowered to suppress power. Plant conditions are:

Reactor water level	-45 inches (N036/N037)
Reactor pressure	800 psig
Drywell pressure	0.6 psig
Suppression pool temp	130°F

Suppression Pool Cooling valves have closed. Returning Suppression Pool Cooling to service requires:

a. placing the Think Switch to Manual only.

b. bypassing the 2/3 core height interlock only.

c. placing the Think Switch to Manual, then bypassing the 2/3 core height interlock.

d. bypassing the 2/3 core height interlock, then placing the Think Switch to Manual.

QUESTION 19 POINT VALUE: 1.00

Consider the following normal RHR suction valve interlocks for control of Reactor water level:

1. Shutdown Cooling suction isolation valves (F008/F009) will automatically isolate on low Reactor water level.

2. Shutdown Cooling pump suction valve (F006) cannot be opened unless Torus common suction valve (F020) is closed.

During Shutdown Cooling operation from Remote Shutdown stations per AOP-32.0, which of the above (if any) interlocks are functional?

a. 1 only.

b. 2 only.

c. both 1 and 2.

d. neither 1 nor 2.

QUESTION 20 POINT VALUE: 1.00

Following a small line break, RHR Loop A is placed in Drywell and Suppression Chamber Spray using required overrides per SEP-02 and SEP-03. Plant conditions are:

Reactor Water Level	+175 inches
Reactor Pressure	900 psig
Drywell Pressure	15.0 psig

Reactor water level drops to -60 inches. How will RHR Loop A and RHR Service Water (RHR SW) Loop A respond?

The RHR Loop A drywell/suppression chamber spray valves:

a. auto close, RHR SW Loop A pump(s) trip.

b. auto close, RHR SW Loop A remains running.

c. remain open, RHR SW Loop A pump(s) trip.

d. remain open, RHR SW Loop A remains running.

QUESTION 21 POINT VALUE: 1.00

Unit Two (2) is operating at 100% power with an active 7 day LCO for RHR Pump 2A being under clearance. An AO reports that the breaker for RHR Room Cooler 2B at MCC 2XB is tripped magnetically.

What action is required by Technical Specifications and by OI-01.08, Control Of Equipment And System Status?

a. Room Coolers are not required for RHR Operability, continue in the active 7 day LCO.

b. Room Coolers are required for RHR Operability, however continue in the active 7 day LCO.

c. Declare RHR Loop B Inoperable and place the unit in HOT SHUTDOWN within 6 hours.

d. Declare RHR Loop B Inoperable and place the unit in HOT SHUTDOWN within 12 hours.

QUESTION 22 POINT VALUE: 1.00

During normal power operation of Unit Two (2), an ECCS Division I Trip Cabinet Trouble alarm is received. Investigation shows that BOTH power supplies to the trip cabinet (XU-63) have failed and all associated trip unit meters indicate downscale with no trip lights lit.

A DBA LOCA then occurs resulting in Reactor water level rapidly dropping below the Top of Active Fuel and rapid Reactor depressuriztion. How will Division I Low Pressure ECCS (Core Spray 2A and RHR LPCI Loop 2A) respond?

a. Core Spray 2A initiates, LPCI 2A fails to initiate.

b. Core Spray 2A and LPCI 2A both fail to initiate.

c. Core Spray 2A fails to initiate, LPCI 2A initiates.

d. Core Spray 2A and LPCI 2A will both auto initiate.

QUESTION 23 POINT VALUE: 1.00

Unit Two (2) HPCI has automatically initiated on a valid initiation signal. The operator observes the following indications:

Steam Supply Pressure	0 psig
Turbine Exhaust Pressure	0 psig
Pump Discharge Pressure	0 psig
Turbine Speed	1000 RPM, lowering
HPCI TURB TRIP SOL ENERG	NOT Alarming

Which of the following would explain the above indications?

a. Isolation due to ruptured exhaust diaphram.

b. Overspeed trip from speed feedback signal failure.

c. Loss of oil pressure to the turbine control system.

d. Loss of 125 VDC input to the 24/52.5 VDC power supplies.

QUESTION 24 POINT VALUE: 1.00

Unit One (1) has experienced a high Reactor pressure transient following a Main Turbine trip. Current plant conditions:

All rods in
Reactor Pressure 950 psig controlled by EHC
Eleven (11) SRV's green indicating lights lit
Eight (8) amber memory lights illuminated

Determine the extent of the pressure transient.

a. 1122 psig

b. 1132 psig

c. 1142 psig

d. 1152 psig

QUESTION 25 POINT VALUE: 1.00

Unit One (1) is operating at power with Core Spray Pump 1B under clearance. A Loss of Off-site Power occurs to BOTH units. Diesel Generators (DGs) 2 and 4 auto start, DGs 1 and 3 trip and lockout.

A stuck open SRV and loss of high pressure injection causes Reactor water level to lower. All available low pressure ECCS pumps have been manually started. Plant conditions are:

Reactor water level	LPCI/Core Spray initiation signal just received
Reactor pressure	650 psig
Drywell pressure	0.6 psig
ADS Inhibit Switches	AUTO

Assuming NO operator action, ADS will:

a. Auto initiate in 1 minute 23 seconds.

b. Auto initiate in 1 minute 45 seconds.

c. Not auto initiate due to lack of high drywell pressure.

d. Not auto initiate due to lack of ECCS pump permissive.

QUESTION 26 POINT VALUE: 1.00

Unit Two (2) has a Loss Of Off-Site Power. HPCI and RCIC both failed. Reactor Water Level dropped to +30 inches before CRD reversed the level trend. Current conditions are:

Reactor water level	+60 inches, rising
Reactor pressure	800-1000 psig using SRVs
Drywell pressure	2.2 psig, rising
Average drywell temp	180°F, rising
Generator primary lockout	Tripped

RBCCW Pumps are tripped, and Nuclear Service Water (NSW) cooling water valves (SW-V103/V106) are closed. What actions are required to restore RBCCW to control containment parameters?

a. Reset the Primary Generator lockout, align RBCCW cooling to the conventional header.

b. Reset the Primary Generator lockout, reopen NSW cooling water valves SW-V103/V106.

c. Reset Core Spray initiation logic, align RBCCW cooling to the conventional header.

d. Reset Core Spray initiation logic, reopen NSW cooling water valves SW-V103/V106.

QUESTION 27 POINT VALUE: 1.00

Unit Two (2) is operating steady state at rated power with the following Electro Hydraulic Control (EHC) conditions:

Reactor pressure	1005 psig
EHC Pressure Setpoint	920 psig
PAM pressure	950 psig
Pressure regulator A	In control
Pressure regulator B	5 psig bias

The Pressure Averaging Manifold (PAM) pressure input to Pressure Regulator A fails low. The PAM pressure input to Pressure Regulator B is unaffected. How will the EHC System respond?

a. Pressure regulator B takes control and stabilizes PAM pressure at 945 psig.

b. Pressure regulator B takes control and stabilizes PAM pressure at 955 psig.

c. Control valves close, reactor pressure and neutron flux rise and the reactor scrams.

d. Control/Bypass valves open and steam line pressure lowers to the Group 1 isolation setpoint.

QUESTION 28 POINT VALUE: 1.00

Unit One (1) is operating at 100% power with the following Condensate system alignment:

Condensate pumps	A & B Running
Condensate pump	C Standby/Auto
Condensate Booster pumps	B & C Running
Condensate Booster pump	A Under clearance

4KV BOP Bus 1D trips and locks out due to a bus fault. The transient results in a Reactor scram. What is the availability of the Condensate system to provide makeup to the Reactor vessel?

Condensate pump:

a. A is available, no Condensate Booster pump is available.

b. B is available, no Condensate Booster pump is available.

c. A is available and Condensate Booster pump C is available.

d. B is available and Condensate Booster pump C is available.

QUESTION 29 POINT VALUE: 1.00

Unit Two (2) is operating at 100% power. The Digital Feedwater Control System (DFCS) is aligned as follows:

Master Controller	Auto, set at 187 inches
Level instrument N004A	187 inches
Level instrument N004B	187 inches
Level instrument N004C	failed downscale
Mode Select switch	3 Element
Level Select switch	Level A

Level instrument N004A fails downscale. Assuming no operator action, Reactor water level will:

a. rise and flood the main steam lines.

b. drop to the low level scram setpoint.

c. rise resulting in a main turbine and feed pump trip.

d. remain at 187" with level instrument N004B in control.

QUESTION 30 POINT VALUE: 1.00

Unit Two (2) is in operation at 100% power. The following annunciator and conditions exist:

A-07, 1-2 RFP FW CONTROL SIGNAL FAILURE
Amber light above A RFPT lockout switch out
Amber light above B RFPT lockout switch lit

A Reactor Recirculation Pump trips. Reactor power lowers and Reactor water level begins to rise uncontrollably. How can the operator restore Reactor water level to the normal band?

a. Operate RFP A MSC control in the Lower direction.

b. Operate RFP B MSC control in the Lower direction.

c. Place RFP A MGU in Manual and lower output demand.

d. Place RFP B MGU in Manual and lower output demand.

QUESTION 31 POINT VALUE: 1.00

The Control Building Ventilation system has initiated in the radiation protection mode. The alignment of system controls is:

Emergency Filtration Fan A	Pref
Emergency Filtration Fan B	Stby

Emergency Filtration Fan A has been running for 10 minutes when a high temperature is sensed in the Train A charcoal bed. Assuming the radiation initiation signal is still present, Emergency Filtration Fan A will:

a. immediately trip and Fan B will immediately auto start.

b. remain running since the high temperature trip is bypassed.

c. immediately trip and Fan B will start after a 10 second delay.

d. remain running since the high temperature provides alarm only.

QUESTION 32 POINT VALUE: 1.00

A Loss of Off-Site Power has occured. Secondary Containment isolated. Reactor Building Ventilation was restarted by guidance of EOP-03-SCCP using SEP-04. Plant conditions are:

Reactor Water Level is +150 inches, slowly rising
Drywell Pressure is 1.2 psig, slowly rising
CAC Vent Purge Isol Ovrd (CAC-CS-5519) is in OVERRIDE
Reactor Building Vent Rad Monitors have been reset
PCIS Isolation Reset push buttons on P601 have been depressed

Which of the following would cause the Reactor Building to re-isolate?

a. Drywell pressure rises above 2.0 psig.

b. Reactor level drops to the Top Of Active Fuel.

c. Main Stack Radiation Monitor exceeds the Hi-Hi setpoint.

d. Reactor Building Vent Exhaust temperature exceeds 140°F.

QUESTION 33 POINT VALUE: 1.00

Following a Loss of Off-Site Power, Diesel Generator #1 (DG1) is running in AUTO, tied to Bus E1. DG1 parameters:

Kilowatt load	3500 KW
Terminal Voltage	4160 Volts
Reactive load	1300 KVAR
Frequency	60 Hz

Off-Site Power has been restored and the BOP buses energized from the SAT. The BOP bus to E1 Master/Slave breaker is still open.

The operator depresses the DG #1 CONTROL ROOM MANUAL push button on the RTGB. DG #1 frequency will be approximately:

a. 57 Hz.

b. 59 Hz.

c. 61 Hz.

d. 63 Hz.

QUESTION 34 POINT VALUE: 1.00

The following sequence of events occurs on Unit One (1):

Time = 0 seconds	Off-site power is lost
Time = 5 seconds	A LOCA signal is recieved
Time = 10 seconds	Diesel generators energize thier respective E Buses

The Motor Driven Fire Pump normal feeder breaker from:

a. Bus E1 closes at Time = 25 seconds.

b. Bus E2 closes at Time = 25 seconds.

c. Bus E1 closes at Time = 30 seconds.

d. Bus E2 closes at Time = 30 seconds.

QUESTION 35 POINT VALUE: 1.00

Unit One (1) is in an outage with the SAT energized and the UAT in backfeed alignment. All backfeed selector switches are in the BACKFEED position. Electrical system alignment:

BOP Buses 1C/1D	Powered from UAT
E Buses E1/E2	Powered from BOP Buses
DGs 1, 2, 3, 4	Operable in standby alignment

A sudden fault pressure occurs in the Main Power Transformer resulting in a Backup Main Generator lockout. How will the electrical distribution system respond?

a. BOP buses 1C/1D transfer to the SAT, four DGs receive an auto start signal.

b. BOP buses 1C/1D transfer to the SAT, no DGs receive an auto start signal.

c. BOP buses 1C/1D are de-energized, four DGs receive an auto start signal.

d. BOP buses 1C/1D are de-energized, DGs 1 and 2 only receive an auto start signal.

QUESTION 36 POINT VALUE: 1.00

Unit One (1) is in a refueling outage with the SAT de-energized and a UAT backfeed in service. Unit Two (2) is operating at 100% power. The following Diesel Generator start times to rated speed are recorded during PT-12.8:

DG1	9.8 seconds
DG2	10.2 seconds
DG3	9.3 seconds
DG4	9.6 seconds

What is the maximum time Unit Two (2) may continue POWER OPERATIONS without entering a Technical Specification Shutdown Statement?

a. 12 hours.

b. 72 hours.

c. 7 days.

d. 45 days.

QUESTION 37 POINT VALUE: 1.00

A "250 V BATT A GROUND" alarm has been received on Unit One (1). The following readings are reported from the Battery Room:

P Bus	0.6 milliamps
N Bus	3.4 milliamps
Charger 1A-1	135 volts, in float
Charger 1A-2	135 volts, in float

Per OP-51 and AI-115, the ground is on the:

a. P Bus, action level 1 applies.

b. P Bus, action level 2 applies.

c. N Bus, action level 1 applies.

d. N Bus, action level 2 applies.

QUESTION 38 POINT VALUE: 1.00

Unit One (1) is operating at 100% power. The UPS system is in its normal alignment for both units. A total loss of UPS occurs on Unit One (1), followed shortly by a spurious Reactor scram. Plant conditions:

Reactor water level lowers to +135 inches
Reactor pressure 950 psig controlled by EHC
APRM recorders on P603 indicate 100% power
Digital Feedwater controller displays are blank

How will the loss of UPS affect the plant during this transient?

a. Reactor power cannot be determined to be less than 3% from P603.

b. Reactor feed pumps will not respond to the reduced Reactor water level.

c. EHC pressure control will be lost as the main turbine coasts down to zero speed.

d. Reactor Building HVAC is lost until the main stack rad monitor is transferred to Unit Two.

QUESTION 39 POINT VALUE: 1.00

An Auxiliary Operator has received 1.95 rem TEDE for the current year. The AO is needed to perform work in a 25 mrem/hr field. The work is expected to last 3 hours.

In accordance with NGGM-PM-0002, Radiation Control and Protection Manual, the worker requires:

a. approval from the Manager - E&RC to exceed the annual administrative dose limit.

b. approval from the Plant General Manager to exceed the annual administrative dose limit.

c. approval from the Site Vice President to exceed the annual administrative dose limit.

d. no special authorizations since the annual administrative limit should not be exceeded.

QUESTION 40 POINT VALUE: 1.00

Both Units have lost off-site power. The only available Diesel Generator is DG2. Buses E2 and E4 cannot be cross-tied. Unit Two (2) UPS has been de-energized for DC load stripping.

What instruments are available to monitor Reactor Water Level on the Unit Two (2) RTGB?

a. Fuel Zone indicator N036 only.

b. Fuel Zone indicator N037 only.

c. Fuel Zone indicator N036 and Narrow Range indicators N004A/B/C.

d. Fuel Zone indicator N037 and Narrow Range indicators N004A/B/C.

QUESTION 41 POINT VALUE: 1.00

You are escorting a visitor with a red badge in the protected area. It becomes desired to temporarily give up your escort duties.

How may this be accomplished?

a. You and the visitor must exit the protected area.

b. You may turn over escort duties to a security guard only.

c. You may turn over escort duties to any other qualified escort, and notify security at the access point you entered.

d. You may turn over escort duties to any other qualified escort, and notify security at the secondary alarm station.

QUESTION 42 POINT VALUE: 1.00

During operation of Unit One (1), a line break in the drywell results in a Reactor Scram. All control rods fully insert and immediate operator actions are completed.

Reactor Water Level rapidly drops off-scale low on the Fuel Zone instruments prior to Low Pressure ECCS initiating and restoring adequate core cooling.

Technical Specifications requires that the NRC Operations Center be notified within one hour, and the:

a. Plant Manager - Brunswick Nuclear Plant within 4 hours.

b. Plant Manager - Brunswick Nuclear Plant within 24 hours.

c. Vice President - Brunswick Nuclear Plant within 4 hours.

d. Vice President - Brunswick Nuclear Plant within 24 hours.

QUESTION 43 POINT VALUE: 1.00

Which one of the following sets of conditions meets the requirements for OPERABILITY of the Standby Liquid Control System for Unit Two (2)?

	TANK LEVEL	SOLUTION CONCENTRATION	SOLUTION TEMPERATURE
a.	3150 gal	16.0%	70°F
b.	3200 gal	14.5%	75°F
c.	3500 gal	13.0%	80°F
d.	3400 gal	15.0%	70°F

QUESTION 44 POINT VALUE: 1.00

A Temporary Procedure Change is developed and designated as Revision To Follow. This Temporary Change receives:

Interim approval on March 2nd
Final approval on March 6th

What is the last date this Temporary change may be used WITHOUT receiving any allowable extention(s)?

a. May 1st.

b. May 5th.

c. May 31st.

d. June 4th.

QUESTION 45 POINT VALUE: 1.00

Which of the following describes the level of use of Emergency Operating Procedures?

a. Reference use procedures.

b. Continuous use procedures.

c. Information use procedures.

d. Exempt from level of use requirements.

QUESTION 46 POINT VALUE: 1.00

Per PLP-21, which of the following would be an UNACCEPTABLE method of performing independent verification by use of RTGB indications?

Independent verification of a Core Spray System:

a. pump breaker closure using red light indication.

b. pump breaker rack in status using green light indication.

c. injection valve opening using system flow indication meter.

d. injection valve standby position using green light indication.

QUESTION 47 POINT VALUE: 1.00

A Station Blackout requires simultaneous execution of the Emergency Operating Procedures, Emergency Plar Procedures and AOP-36.2.

The Work Control Center SRO reports to the Control Room functioning in a dual role as the Shift Technical Advisor (STA).

Per OI-01.01, since the STA holds an active license, the STA may be allowed to:

a. function as the Site Emergency Coordinator.

b. install circuit alterations required by the EOPs.

c. silence and acknowledge, but not reset annunciators.

d. direct other operators in AOP-36.2 activities while the Unit SCO directs EOPs.

QUESTION 48 POINT VALUE: 1.00

During RTGB walkdown in preparation for shift turnover, the oncoming Reactor Operator notes that annunciator A-5 2-2, Rod Out Block:

Alarm is sealed in
Has a yellow dot affixed to the window

If the signal causing this alarm clears, then comes back in without the alarm being reset, the subsequent alarm condition is indicated by the alarm window flashing:

a. slowly with an audible alarm.

b. rapidly with an audible alarm.

c. slowly, then rapidly without an audible alarm.

d. rapidly, then slowly without an audible alarm.

QUESTION 49 POINT VALUE: 1.00

A plant shutdown is in progress per GP-05 for refueling. Current plant conditions are:

Reactor power 20%
Drywell oxygen 19.0%

Drywell entry may be made with the approval of the Manager-E&RC and the authorization of the Shift Superintendent only if reactor power is reduced at least:

a. 5%, oxygen concentration is acceptable.

b. 5%, oxygen concentration must be raised.

c. 15%, oxygen concentration is acceptable.

d. 15%, oxygen concentration must be raised.

QUESTION 50 POINT VALUE: 1.00

Which of the following valve/actuator types may be used as a clearance boundary isolation component, provided the associated restrictions of AI-58 are satisfied?

a. Solenoid operated ball valve marked as fail-closed on the print.

b. Motor operated globe valve normally used as a flow control valve.

c. Pressure balanced diaphragm operated pinch valve not equipped with a handwheel.

d. Double acting cylinder operated butterfly valve not marked as fail-closed on the print.

QUESTION 51 POINT VALUE: 1.00

A clearance request has been received for a fluid system with the following normal operating parameters:

Pressure	475 psig
Temperature	175°F

Per AI-58, the clearance:

a. May use single valve boundary isolation.

b. Must use dual valve boundary isolation due to pressure only.

c. Must use dual valve boundary isolation due to temperature only.

d. Must use dual valve boundary isolation due to pressure and temperature.

QUESTION 52 POINT VALUE: 1.00

During accident conditions on Unit One (1), the following plant conditions exist:

Reactor water level	-70 inches (Fuel Zone)
Reactor pressure	1100 psig
Drywell average temp	190°F
Drywell ref leg area temp	270°F
Injection sources	None available

Under these conditions, peak fuel clad temperature will not exceed:

a. 1500°F, provided Reactor water level remains above -80 inches.

b. 1500°F, provided Reactor water level remains above -90 inches.

c. 1800°F, provided Reactor water level remains above -80 inches.

d. 1800°F, provided Reactor water level remains above -90 inches.

QUESTION 53 POINT VALUE: 1.00

Following accident conditions, the crew is executing the Reactor Vessel Flooding Procedure, EOP-01-RXFP. Plant conditions are:

Control rods	Fully inserted
Reactor water level	Unknown

The operator is directed to control injection flow to the Reactor to maintain at least _____ SRV/ADS Valves open and Reactor pressure:

a. 4; above the Minimum Alternate Flooding Pressure.

b. 5; above the Minimum Alternate Flooding Pressure.

c. 4; at least 50 psig above suppression chamber pressure.

d. 5; at least 50 psig above suppression chamber pressure.

QUESTION 54 POINT VALUE: 1.00

A heavy influx of marsh grass on the Circulating Water Screens has caused a loss of all Circulating Water pumps and a reactor scram. Plant conditions are:

Group 1 isolated
Condenser vacuum is 0" Hg
Turbine speed is 500 rpm, dropping
EHC Electrical Malfunction in alarm due to loss of the PMG.

The marsh grass is now cleared and the Circulating Water System has been restarted. Is the Main Condenser available as a heat sink?

a. No, the MSIVs are closed.

b. No, the EHC system is not available.

c. No, the condenser is not under vacuum.

d. Yes, all required systems are available.

QUESTION 55 POINT VALUE: 1.00

A Reactor scram occurs on low Reactor water level following a loss of feedwater. The following sequence occurs:

0100	Reactor scrams
0115	Reactor Vessel Control Procedure is enterred
0130	Circuit alteration installed per RVCP for HPCI
0145	EOPs are exited
0200	HPCI alteration restored per SEP-08

The plant is at 900 psig with the Mode Switch in SHUTDOWN. The SRO should initiate a WR/JO to functionally test the altered HPCI circuit and initiate:

a. an active LCO with a start time of 0130.

b. an active LCO with a start time of 0145.

c. a tracking LCO with a start time of 0130.

d. a tracking LCO with a start time of 0145.

QUESTION 56 POINT VALUE: 1.00

Following a loss of all high pressure injection on Unit One (1), seven ADS valves have been manually opened to restore adequate core cooling. Plant conditions are now:

Reactor water level	+25", N026A/B
Reactor water level	+140", N036/37
Reactor pressure	25 psig
Drywell average temp	155°F
Drywell ref leg area temp	280°F

Reactor water level may be determined using:

a. N026A/B only.

b. N036/37 only.

c. Both N026A/B and N036/37.

d. Neither N026A/B nor N036/37.

QUESTION 57 POINT VALUE: 1.00

A Unit Two (2) reactor scram has occurred. Seven control rods failed to fully insert and are between positions 08 and 18. Conditions are:

All APRM Downscale lights are LIT
MSIVs are open
Total Steam Flow 3.6 E6 lbm/Hr, dropping
Reactor Pressure 900 psig, dropping
Narrow Range Level Instruments (N004s) +155 inches, rising
Master Feedwater setpoint at +170"
Two Reactor Feed Pumps in operation

With current plant conditions, the operator is required as an <u>IMMEDIATE</u> action to:

a. trip the Main Turbine.

b. trip one Reactor Feed Pump.

c. place the Mode Switch to SHUTDOWN.

d. enter Alternate Control Rod Insertion.

QUESTION 58 POINT VALUE: 1.00

Following a group 1 isolation and a reactor scram, the operating crew is performing the Reactor Scram Procedure, EOP-01-RSP. Plant conditions are:

Reactor water level	195 inches, slowly rising (N004s)
Reactor pressure	800-1000 psig, controlled by SRVs
Drywell pressure	1.0 psig, slowly rising
Suppression pool temp	94°F, slowly rising
Suppression pool level	-27.5", slowly rising

The operating crew is <u>required</u> to enter EOP-01-RVCP and execute concurrently with the scram procedure if:

a. drywell pressure rises to 1.5 psig.

b. reactor water level rises to +230 inches.

c. suppression pool temperature rises to 111°F.

d. suppression pool level rises to -26.5 inches.

QUESTION 59 POINT VALUE: 1.00

Following a Unit One (1) Reactor scram, the crew has entered and is executing EOP-01-RSP, Reactor Scram Procedure. Plant conditions are:

Reactor water level	220", rising
Reactor pressure	945 psig, stable
MSIVs	Open

Per EOP-01-RSP, the MSIVs must be manually closed if Reactor water level cannot be maintained below:

a. 230"

b. 240"

c. 250"

d. 260"

QUESTION 60 POINT VALUE: 1.00

The entry conditions for Unit One (1) EOP-01-RVCP, Reactor Vessel Control Procedure for Reactor pressure and water level are Reactor pressure is greater than:

a. 1035 psig, or Reactor water level less than 153".

b. 1035 psig, or Reactor water level less than 166".

c. 1060 psig, or Reactor water level less than 153".

d. 1060 psig, or Reactor water level less than 166".

QUESTION 61 POINT VALUE: 1.00

During an ATWS on Unit One (1), all Bypass Valves were full open together with one SRV for pressure control. Conditions required injection to the Reactor vessel to be terminated and prevented.

Current conditions are:

Reactor power	22%, dropping
Reactor water level	+125 inches, dropping
Reactor pressure	952 psig
Suppression pool temp	115 °F, dropping
Drywell pressure	0.5 psig, dropping
Bypass valves	3 1/2 open
SRVs	All Closed
SLC tank level	66%

What action is required?

a. Continue to lower level until Reactor power is below 3% or Reactor water level reaches TAF.

b. Continue to lower level until Reactor power is below 3% or Reactor water level reaches LL4.

c. Establish a level band no higher than +125 inches and no lower than TAF.

d. Establish a level band no higher than +125 inches and no lower than LL4.

QUESTION 62 POINT VALUE: 1.00

During an ATWS on Unit Two (2), a Safety Relief Valve fails open. The following plant conditions exist:

Reactor Power	APRMs downscale
Reactor water level	-60 inches (Fuel zone)
Reactor pressure	700 psig, lowering
Drywell ref leg area temp	187°F
HPCI	NOT available
RCIC, CRD, SLC	Injecting

Assume Reactor water level remains constant. The Reactor must be Emergency Depressurized if Reactor pressure drops to:

a. 300 psig, RCIC may inject during the depressurization.

b. 600 psig, RCIC may inject during the depressurization.

c. 300 psig, RCIC must be terminated prior to opening ADS valves.

d. 600 psig, RCIC must be terminated prior to opening ADS valves.

QUESTION 63 POINT VALUE: 1.00

Following an incomplete Reactor scram, the operating crew is executing EOP-01-LPC, Level/Power Control. A decision step is reached asking "Is The Reactor Shutdown?".

Which of the following conditions would satisfy the definition of "SHUTDOWN" as it applies to the Reactor?

a. All operable APRMs indicate downscale.

b. The Reactor is subcritical on range 6 of IRMs.

c. The entire SLC Tank has been injected to the Reactor.

d. Hot Shutdown Boron Weight has been injected to the Reactor.

QUESTION 64 POINT VALUE: 1.00

During severe accident conditions on Unit Two (2), the following sequence of events occurs:

0100	Reactor scrams
0630	Reactor water level undetermined
0700	Reactor flooding conditions initially established
0930	Minimum core flooding interval is satisfied
1600	Injection to the Reactor is terminated

At what time must injection be re-established if Reactor water level cannot be determined?

a. 1604

b. 1605

c. 1606

d. 1607

QUESTION 65 POINT VALUE: 1.00

Following a reactor scram and a group 1 isolation, SRVs are being used to maintain reactor pressure 900-1000 psig.

Which of the following conditions requires ALL group 1 isolations to be defeated and the reactor vessel rapidly depressurized to the main condenser?

a. Suppression Pool Level is +4' 6"
Suppression Pool Temperature is 95°F

b. Suppression Pool Level is -1' 6"
Suppression Pool Temperature is 170°F

c. Suppression Pool Level is -4' 3"
Suppression Pool Temperature is 156°F

d. Suppression Pool Level is -8' 1"
Suppression Pool Temperature is 105°F

QUESTION 66 POINT VALUE: 1.00

Following a large Recirculation line rupture, EOP-01-PCFP, Primary Containment Flooding Procedure, is being exectuted. The following indications are available:

CAC-LI-2601-1	+5.9 feet
CAC-PI-1257-2A	23 psig
CAC-PI-1230	21 psig
CAC-PI-4176	25 psig
CAC-PR-1257-1	22 psig

What is Primary Containment water level?

a. +14.5 feet

b. +9.9 feet

c. +7.6 feet

d. +4.6 feet

QUESTION 67 POINT VALUE: 1.00

During accident conditions, the operating crew is executing the Primary Containment Flooding Procedure, EOP-01-PCFP.

This procedure requires that the Inboard Steam Line Drain Isolation Valve (B21-F016) be disabled (breaker opened) in the:

a. open position prior to Primary Containment water level reaching 21 feet.

b. open position prior to Primary Containment water level reaching 23 feet.

c. closed position prior to Primary Containment water level reaching 21 feet.

d. closed position prior to Primary Containment water level reaching 23 feet.

QUESTION 68 POINT VALUE: 1.00

A seismic event has occurred that has resulted in a Loss of Off-Site Power and high power ATWS conditions.

The SLC Storage Tank outlet line completely severed at the tank during the earthquake. The SLC tank is <u>EMPTY</u> making the SLC pumps unavailable for boron injection.

Which system should be selected for alternate boron injection?

a. CRD

b. RCIC

c. RWCU

d. Condensate

QUESTION 69 POINT VALUE: 1.00

A Condensate header rupture in the cable spread area of the Control Building has resulted in a loss of all UPS and RPS power.

Plant status is as follows:

Blue scram lights	137 illuminated
IRM Indications	50 on Range 10

What method of EOP-01-LEP-02, Alternate Control Rod Insertion, would be <u>MOST</u> effective in inserting the withdrawn rods?

a. Vent the scram air header.

b. Vent the overpiston area of control rods.

c. Scram individual rods with the scram test switches.

d. Insert control rods with the Reactor Manual Control System.

QUESTION 70 POINT VALUE: 1.00

During a low reactor water level condition, Alternate Coolant Injection using demineralized water is being aligned using the HPCI system. A valid HPCI isolation signal is present, resulting in an automatic closure signal to the HPCI Injection valve (E41-F006).

How is the HPCI Injection Valve (E41-F006) opened to provide injection to the reactor?

E41-F006 is opened from the:

a. RTGB after placing the HPCI ASSD Interlock Defeat Switch on the RTGB to BYPASS.

b. MCC by placing the breaker's NORMAL/LOCAL switch to LOCAL to bypass valve interlocks.

c. RTGB and the breaker at the MCC is opened by an AO when the valve indicates full open.

d. RTGB after jumpers are installed to bypass the valve auto closure interlocks.

QUESTION 71 POINT VALUE: 1.00

Following a loss of drywell cooling, a small steam leak in the drywell results in the following containment conditions:

Drywell pressure	9 psig, rising
Suppression chamber pressure	8 psig, rising
Suppression pool level	+2 feet
Average Drywell temp	270°F, rising

The crew is directed to initiate drywell spray to control drywell temperature. Under current plant conditions, drywell spray may:

a. be initiated, all required conditions are met.

b. NOT be initiated, suppression pool level is too high.

c. NOT be initiated, suppression chamber pressure is too low.

d. NOT be initiated, conditions are in the UNSAFE region of the Drywell Spray Initiation Limit.

QUESTION 72 POINT VALUE: 1.00

The Suppression Chamber Spray Initiation Pressure is ________ in the Suppression Chamber and is based on:

a. 2.7 psig; intrusion of air into primary containment due to Reactor Building-Torus vacuum breaker operation.

b. 2.7 psig; the lowest suppression chamber pressure that RHR system logic will allow sprays to be initiated.

c. 13 psig; 95% of the noncondensibles in the drywell have been transferred to the suppression chamber airspace.

d. 13 psig; the highest pressure that initiation of sprays will prevent exceeding the Pressure Suppression Pressure Limit.

QUESTION 73 POINT VALUE: 1.00

During an accident on Unit One (1), the following primary containment and plant conditions exist:

Reactor pressure	798 psig
Suppression pool level	-42 inches
Suppression pool temperature	171°F
Suppression chamber pressure	17 psig

Current conditions are in the:

a. SAFE region of all Containment Limits.

b. UNSAFE region of the Heat Capacity Level Limit.

c. UNSAFE region of the Heat Capacity Temperature Limit.

d. UNSAFE region of the Pressure Suppression Pressure Limit.

QUESTION 74 POINT VALUE: 1.00

During accident conditions on Unit Two (2), plant conditions are:

Reactor level	Below TAF
Reactor pressure	500 psig
HPCI and RCIC	injecting at rated flow
Suppression pool level	-6.6 ft
Drywell Temperature	310°F (Average)
Drywell H2 concentration	5.1% (Compensated)
Drywell O2 concentration	6.1% (Compensated)

The operator should perform Emergency Depressurization and:

a. Vent the drywell irrespective of radiation release.

b. Initiate drywell sprays irrespective of adequate core cooling.

c. Vent the suppression chamber irrespective of radiation release.

d. Terminate HPCI injection irrespective of adequate core cooling.

QUESTION 75 POINT VALUE: 1.00

A primary system discharging into Secondary Containment has resulted in one area exceeding the Maximum Safe Operating Radiation Level, but within the EQ envelop. The radiation level in this area is subsequently reduced below the Maximum Safe value.

A second area subsequently exceeds its Maximum Safe Operating Radiation Level. What action is required by the Secondary Containment Control Procedure?

a. Shutdown the Reactor per GP-05.

b. Scram the Reactor and initiate a cooldown ≤100°F/Hour.

c. Scram the Reactor and initiate a cooldown >100°F/Hour.

d. Scram the Reactor and open seven ADS valves.

QUESTION 76 POINT VALUE: 1.00

While performing PT 9.2, HPCI OPERABILITY TEST, the HPCI steam supply line ruptured. HPCI failed to automatically isolate and attempts to manually isolate HPCI are unsuccessful.

The following Steam Leak Detection NUMAC channels are in alarm:

B21-XY-5949A, Channel A3-3, reading 303°F
B21-XY-5949B, Channel A3-3, reading 298°F
B21-XY-5948A, Channel A5-1, reading 301°F
B21-XY-5948B, Channel A5-1, reading 296°F

No other channels are in alarm. What action is <u>required</u> to be be taken?

a. Scram the reactor and commence a cooldown at normal rates.

b. Shutdown the reactor using GP-05 or scram the reactor as directed by the Shift Supervisor.

c. Scram the reactor and emergency depressurize.

d. Scram the reactor and rapidly depressurize to the main condenser.

QUESTION 77   POINT VALUE: 1.00

Unit Two (2) is operating at power when a rupture of RWCU piping downstream of the Non Regenerative Heat Exchangers occurs. RWCU Inboard Isolation Valve (G31-F001) and Outboard Isolation Valve (G31-F004) BOTH fail in the open position. Plant conditions:

Rx Bldg 50' temp	135°F
Rx Bldg 20' temp	105°F
S Core Spray Room	Flood Level Hi Hi alarm sealed in
S RHR Room	Flood Level Hi alarm sealed in

The operating crew is required to enter EOP-03-SCCP and:

a. continue attempts to isolate the leak, commence an immediate plant shutdown per GP-05.

b. continue attempts to isolate the leak, scram the reactor when 50' temperature exceeds 140°F.

c. immediately scram the reactor and consider anticipation of emergency depressurization.

d. immediately scram the reactor and open seven ADS valves for emergency depressurization

QUESTION 78 POINT VALUE: 1.00

Following a small line break in the drywell HPCI/RCIC auto initiate. A HPCI steam leak in the HPCI steam tunnel results in an isolation of HPCI. MSIVs are closed and Feed Pumps are NOT available. Plant conditions are:

Reactor Water Level	90 inches, steady
Drywell Pressure	15 psig
RCIC Flow	500 gpm
RCIC Steam Tunnel	195°F, steady
Reactor Bldg Vent Rad Hi	Alarm cleared

The 30 minute RCIC steam tunnel leak detection isolation timer is running. What required operator action could maintain RCIC availability?

a. Restart Reactor Building Ventilation per SEP-04.

b. Restart Reactor Building Ventilation per OP-37.1.

c. Turn power off to steam leak detection NUMAC modules.

d. Install a circuit alteration to defeat steam leak detection.

QUESTION 79 POINT VALUE: 1.00

Following core damage, an unisolable steam leak in the Turbine Building requires declaration of a General Emergency due to loss of three out of three fission product barriers.

The crew is executing EOP-04-RRCP, Radiation Release Control Procedure. Field surveys and Off-Site dose projections (PEP-03.4.7) are being performed.

When, per EOP-04-RRCP, is Emergency Depressurization of the Reactor required to be initiated?

a. Immediately since a General Emergency has been declared.

b. The Noble Gas release rate reaches 1200% of the Tech Spec limit.

c. The measured dose rate at the site boundary is reported at 110 mrem/hour.

d. Dose projections estimate a Off-Site dose of 4.9 rem thyroid (CDE).

QUESTION 80 POINT VALUE: 1.00

A chemistry sample has been directed due to steadily rising SJAE rad monitor readings. The following sequence of events occur at the times noted:

0900	chemistry reports coolant activity of 4.3 μci/ml based on sample
0905	the SS enters a 12 hour to Hot Shutdown LCO based on coolant activity
0910	the SS declares an Unusual Event based on abnormal core conditions
0915	A plant shutdown is started to comply with Technical Specifications

The NRC must be notified of events in progress no later than:

a. 1000

b. 1005

c. 1010

d. 1015

QUESTION 81 POINT VALUE: 1.00

A General Emergency has been declared on July 4th due to an unisolable RCIC steam line break with indications of fuel failure. The following weather data is availabl·:

Temperature	97°F
Upper wind speed	6.7 mph
Lower wind speed	5.8 mph
Upper wind direction	236.8°
Lower wind direction	237.3°
Stability class	D

Maximum projected off-site dose per PEP-03.4.7 is 45 mrem TEDE and maximum off-site field survey readings are 28 mrem/hour. The release is expected to drop rapidly due to the emergency depressurization of the Reactor in progress.

What Protective Action Recommendation (PAR) should be made to Off-Site Agencies?

a. Evacuate zones A, B, C, G, H, K and shelter zones D, E, F.

b. Evacuate zones A, B, C and shelter zones D, E, F, G, H, K.

c. Shelter all zones due to the release being below EPA Protective Action Guidelines.

d. Shelter all zones since an evacuation is expected to take over eight hours to complete.

QUESTION 82 POINT VALUE: 1.00

A Unit Two (2) Reactor startup is in progress per GP-02. Heatup and pressurization of the Reactor is being performed.

The operating CRD Pump trips. Attempts to restart CRD per OP-08 and AOP-02.0 are unsuccessful.

AOP-02.0 requires the operator to insert a manual Reactor scram only if Reactor pressure is below:

a. 200 psig

b. 400 psig

c. 600 psig

d. 800 psig

QUESTION 83 POINT VALUE: 1.00

Unit One (1) is operating at 100% power when Recirculation Pump 1B trips, resulting in the following conditions:

Total Core Flow (P603)	39 Mlbm/Hour
Total Core Flow (U1CPWTCF)	35 Mlbm/Hour
Indicated Core Plate DP	4.7 psid
APRMs	68%
LPRM Upscale/Downscale alarms	None

What region of the Thermal Power Limitations Map is the plant operating in, and what operator action is required to be taken?

a. Region B, raise total core flow.

b. Region B, insert control rods per ENP-24.

c. Region A, immediately insert a manual scram.

d. 5% Buffer, increase monitoring of nuclear instrumentation.

QUESTION 84 POINT VALUE: 1.00

Unit Two (2) was operating at power when a trip and lockout of BOP bus 2B required the operator to insert a manual Reactor scram. Shortly following the scram, the following indications are noted:

Recirc pump A #1 seal pressure	1000 psig
Recirc pump A #2 seal pressure	1000 psig
Recirc pump B #1 seal pressure	100 psig
Recirc pump B #2 seal pressure	50 psig
Drywell pressure	1.4 psig, rising
Average drywell temp	140°F, rising
Average primary containment temp	126°F, rising

The operator is required to enter:

a. AOP-14.0 and isolate Recirc pump A.

b. AOP-14.0 and isolate Recirc pump B.

c. EOP-02-PCCP and isolate Recirc pump A.

d. EOP-02-PCCP and isolate Recirc pump B.

QUESTION 85 POINT VALUE: 1.00

A situation arises requiring immediate evacuation of the control room prior to completion of any immediate actions per AOP-32.0. RPS is aligned:

RPS Bus A	Powered from RPS MG Set A
RPS Bus B	Powered from RPS MG Set B

If the RPS EPA breakers are opened in the exact sequence specified by AOP-32.0, opening which EPA breaker will result in a reactor scram?

a. EPA Breaker 1.

b. EPA Breaker 2.

c. EPA Breaker 3.

d. EPA Breaker 4.

QUESTION 36 POINT VALUE: 1.00

Following a loss of feedwater on Unit Two (2), HPCI and RCIC are being used to restore Reactor water level to the normal band. The operator notes the following alarms and indications:

250 Batt B Under Voltage	Alarm sealed in
Battery Bus 2B-1 Voltage	0 volts (XU-2)
Battery Bus 2B-2 Voltage	0 volts (XU-2)
Battery Bus 2B-1 Voltage	0 volts (ERFIS)
Battery Bus 2B-2 Voltage	0 volts (ERFIS)

How is the operation of HPCI and RCIC affected by the power loss?

a. HPCI continues to inject to the Reactor, RCIC isolates due to loss of isolation logic power.

b. RCIC continues to inject to the Reactor, HPCI isolates due to loss of isolation logic power.

c. HPCI continues to inject to the Reactor, RCIC coasts down due to loss of flow controller power.

d. RCIC continues to inject to the Reactor, HPCI coasts down due to loss of flow controller power.

QUESTION 87 POINT VALUE: 1.00

Following a Loss of Off-Site Power to Unit One (1), the operator is performing AOP-36.1. Plant conditions are:

Diesel Generator 1	Running at 3575 KW load
Diesel Generator 2	Running at 3680 KW load
Reactor Building HVAC	Isolated

The operator is directed to restart Reactor Building HVAC using three (3) supply fans (75 KW each) and three (3) exhaust fans (45 KW each).

How will starting <u>two</u> supply and exhaust fans from MCC 1XG and <u>one</u> supply and exhaust fan from MCC 1XH affect Diesel Generator (DG) maximum loading?

a. DG1 only maximum load will be exceeded.

b. DG2 only maximum load will be exceeded.

c. DG1 and DG2 maximum load will be exceeded.

d. DG1 and DG2 will remain within maximum load limits.

QUESTION 88 POINT VALUE: 1.00

While operating at rated power, the following indications are noted:

Gen Bus Under Freq Relay	Alarm sealed in
Generator frequency	59.2 Hz

If frequency remains at this value for five minutes, the turbine must be tripped to prevent damage due to excessive:

a. volts/hertz in the main generator windings.

b. volts/hertz in the main transformer windings.

c. resonance vibrations in low pressure turbine blading.

d. resonance vibrations in high pressure turbine blading.

QUESTION 89 POINT VALUE: 1.00

Following a total loss of RBCCW, several control rods fail to fully insert on a manual reactor scram.

How long can the CRD Pumps be operated without Cooling Water per the Abnormal Operating Procedure?

a. 40 minutes.

b. 30 minutes.

c. 20 minutes.

d. 10 minutes.

QUESTION 90 POINT VALUE: 1.00

Unit Two (2) was at 20% power during plant startup when a sudden rise in off gas flow is accompanied by a lowering condenser vacuum. The reactor was manually scrammed. Plant conditions:

Condenser vacuum	15" Hg, slowly lowering
Reactor pressure	921 psig, steady
Bypass valves	One partially open

What is the minimum additional reduction in condenser vacuum that would result in the loss of automatic Reactor pressure control?

a. 5" Hg.

b. 6" Hg.

c. 7" Hg.

d. 8" Hg.

QUESTION 91 POINT VALUE: 1.00

Unit One (1) is performing refueling operations. A fuel handling accident results the the following radiation alarms:

Area Rad Refuel Floor Hi (white alarm)
Area Rad Rx Bldg Hi (red alarm, blue bar)
Area Rad Control Room Hi (red alarm, red bar)
Rx Bldg Vent Rad Hi (red alarm, blue bar)
Rx Bldg Vent Rad Hi Hi (red alarm, blue bar)

How will the Reactor Building HVAC and Control Building Emergency Air Filtration (CBEAF) systems respond to the above radiation alarms?

a. Reactor Building HVAC isolates, CBEAF remains in standby.

b. Reactor Building HVAC remains in operation, CBEAF initiates.

c. Reactor Building HVAC remains in operation, CBEAF remains in standby.

d. Reactor Building HVAC isolates and CBEAF initiates.

QUESTION 92 POINT VALUE: 1.00

During normal full power operation of Unit Two (2), the following alarms and indications are noted:

Air Compressor D Trip	Alarm sealed in
Air Compressors A/B/C	Running
Instrument Air Pressure low	Alarm sealed in
Instrument Air header pressure	100 psig

The operator should verify that air compressors A, B and C are loaded and that:

a. Service air isolation valves, PV-706-1 and PV-706-2, have automatically closed.

b. Interruptible air isolation valves, PV-722-1 and PV-722-2, have automatically closed.

c. Standby reactor building air compressors have automatically started and loaded.

d. Backup nitrogen rack isolation valves, RNA-SV-5482 and SV-5481, have automatically opened.

QUESTION 93 POINT VALUE: 1.00

Following a loss of shutdown cooling, Alternate Shutdown Cooling has been established per AOP-15.0. Plant conditions are:

RHR Loop A in suppression pool cooling
RHR Loop B injecting to the Reactor Vessel
Reactor pressure is 115 psig
SRV G is open

It becomes desired to make a slight adjustment to raise the cooldown rate. This may be accomplished by closing SRV G and opening:

a. SRV H.

b. SRV J.

c. SRV K.

d. SRV L.

QUESTION 94 POINT VALUE: 1.00

Unit Two (2) is operating with the following plant conditions:

Reactor power	85%
Core flow	51 Mlbm/hr
Rod line	110%
Recirc MG sets	Scoop tubes locked

Which of the following conditions authorizes the operator to manually initiate Select Rod Insert (SRI)?

a. Condenser vacuum lowers and approaches the turbine trip setpoint.

b. Feedwater heating is partially lost and APRMs approach the scram setpoint.

c. A reactor feed pump trips and reactor level approaches the scram setpoint.

d. A recirculation pump trips placing the plant in Region A of the Thermal Power Limitations Map.

QUESTION 95 POINT VALUE: 1.00

A clorine release of 25 lbs Cl/sec is in progress. The following meteorological data is available:

Air temperature	46.8°F
Lower wind direction	238.7°
Lower wind velocity	7.8 mph
Stability class	E

The concentration of clorine is expected to drop below 10,000 ppm at a downwind distance of between:

a. 0.25 and 0.5 miles.

b. 0.5 and 0.75 miles.

c. 0.75 and 1.0 miles.

d. 1.0 and 1.25 miles.

QUESTION 96 POINT VALUE: 1.00

Unit One (1) is in a Station Blackout. E Buses are being cross-tied. The Reactor is being cooled down at 100°F/Hour. SRVs are being used to reduce Reactor pressure and HPCI/RCIC are being used to maintain Reactor water level.

The following Drywell Temperature readings are reported from the Remote Shutdown Panel:

CAC-TR-778, Point 1	314°F
CAC-TR-778, Point 3	300°F
CAC-TR-778, Point 4	297°F

What action is required?

a. Immediately open seven ADS valves.

b. Raise the cooldown rate to >100°F/hour.

c. Align LPCI for injection, then open seven ADS valves.

d. Align fire water for injection, then open seven ADS valves.

QUESTION 97 POINT VALUE: 1.00

During Station Blackout conditions, outside air temperature drops below 32°F. HPCI is being used to maintain Reactor Water Level above TAF. AOP-36.2 directs HPCI suction valve breakers to be turned OFF.

This action will maintain HPCI suction from the:

a. Suppression Pool to prevent exceeding minimum reactor vessel feedwater nozzle temperature requirements.

b. Suppression Pool to prevent loss of HPCI due to excessive cooling of the pump and turbine lube oil.

c. CST to prevent inadequate flow causing loss of CST suction capability due to suction line freezing.

d. CST to prevent a false low CST level suction transfer signal due to the level switches freezing.

QUESTION 98 POINT VALUE: 1.00

The Control Room has been evacuated due to a fire in the Control Building. Shutdown from outside the Control Room is in progress per ASSD Procedures.

Reactor cooldown is in progress in preparation to place shutdown cooling in service. The following Reactor pressures are recorded at the indicated times:

0000	750 psig
0015	640 psig
0030	390 psig
0045	340 psig
0100	310 psig

The Reactor Vessel cooldown rate is:

a. less than 100°F/Hour, maintain present cooldown rate.

b. greater than 100°F/Hour, maintain present cooldown rate.

c. less than 100°F/Hour, increase cooldown rate to greater than 100°F/Hour.

d. greater than 100°F/Hour, decrease cooldown rate to less than 100°F/Hour.

QUESTION 99 POINT VALUE: 1.00

Core defueling is in progress. All control rods are fully inserted into the reactor core. A fuel assembly has just been placed in the fuel pool and unlatched. The main hoist has been raised to a safe elevation to pass through the cattle chute (**NOT** normal-up) with the bridge still over the fuel pool location.

The next step requires that a fuel assembly be removed from the reactor core and placed in the fuel pool.

When will the ROD BLOCK INTERLOCK #1 light on the Interlock Status Display Panel first light as the next step is performed?

a. As the bridge is moved near the reactor core (LS1 is actuated).

b. When the bridge is over the reactor core (LS1 is actuated) and the main hoist is lowered into the reactor vessel.

c. When the fuel assembly is latched, with both grapple hooks closed.

d. When the fuel assembly is being raised and the main hoist loaded signal is actuated.

QUESTION 100 POINT VALUE: 1.00

Core Spray Pump 2A 4160 volt breaker is racked in per OP-50, with 125V DC available at the switchgear. A LOCA results in a condition requiring the auto start of the pump.

Refer to LL-09113 sheet 15, Core Spray Pump 2A Control Wiring Diagram. The pump breaker is closed by energizing the:

a. X coil, when relay K12A is de-energized and relay K15A is energized.

b. Y coil, when relay K12A is de-energized and relay K15A is energized.

c. X coil, when relay K12A is energized and relay K15A is de-energized.

d. Y coil, when relay K12A is energized and relay K15A is de-energized.

** END OF "NRC 97-1 SRO, Rev 0" EXAMINATION **

NAME: SRO Answer Key DATE: / / SCORE: ______

GRADED BY: ______ ALTERNATE GRADER: ______ (if required)

EXAM: NRC 97-1 SRO, Rev 0 CLASS: HLC 96-1 COURSE CODE: ROA02B

1. A B C **D**	21. A **B** C D	41. A B **C** D	61. A B C **D**
2. A **B** C D	22. A B C **D**	42. A B C **D**	62. **A** B C D
3. A B **C** D	23. A B C **D**	43. A B C **D**	63. A **B** C D
4. A B C **D**	24. A B **C** D	44. **A** B C D	64. A B C **D**
5. A B **C** D	25. **A** B C D	45. A B C **D**	65. A B C **D**
6. A B **C** D	26. A B **C** D	46. A **B** C D	66. A **B** C D
7. A B C **D**	27. A **B** C D	47. A B **C** D	67. **A** B C D
8. **A** B C D	28. A B C **D**	48. A B C **D**	68. A B **C** D
9. A B **C** D	29. **A** B C D	49. A **B** C D	69. A **B** C D
10. A B **C** D	30. A **B** C D	50. A B C **D**	70. A B **C** D
11. A B C **D**	31. **A** B C D	51. **A** B C D	71. A **B** C D
12. **A** B C D	32. A B C **D**	52. A B **C** D	72. A B **C** D
13. A B C **D**	33. **A** B C D	53. A B C **D**	73. A **B** C D
14. A **B** C D	34. A B C **D**	54. A B C **D**	74. A B C **D**
15. **A** B C D	35. A **B** C D	55. A **B** C D	75. A **B** C D
16. A **B** C D	36. **A** B C D	56. **A** B C D	76. **A** B C D
17. A **B** C D	37. **A** B C D	57. **A** B C D	77. A B **C** D
18. **A** B C D	38. A B **C** D	58. A B **C** D	78. **A** B C D
19. A **B** C D	39. A **B** C D	59. A B **C** D	79. A B C **D**
20. A B **C** D	40. A B **C** D	60. A B C **D**	80. A B **C** D

NAME: SRO Answer Key DATE: / / SCORE: ____

GRADED BY: ____ ALTERNATE GRADER: ____ (if required)

EXAM: NRC 97-1 SRO, Rev 0 CLASS: HLC 96-1 COURSE CODE: ROA02B

81. A B C D
82. A B C D
83. A B C D
84. A B C D
85. A B C D
86. A B C D
87. A B C D
88. A B C D
89. A B C D
90. A B C D
91. A B C D
92. A B C D
93. A B C D
94. A B C D
95. A B C D
96. A B C D
97. A B C D
98. A B C D
99. A B C D
100. A B C D

Master
NRC

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: II
Date: 04/25/97	Facility/Unit: Brunswick/1 & 2
License Level: Reactor Operator	Reactor Type: GE BWR-4
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected four hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

WRITTEN EXAMINATION GUIDELINES

1. 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.

2. To pass the examination, you must achieve a grade of 80.00 percent or greater. Every question is worth one point.

3. For an initial examination, the time limit for completing the examination is four hours.

4. You may bring pens and calculators into the examination room. Use only black ink to ensure legible copies.

5. Print your name in the blank provided on the examination cover sheet and the answer sheet. You may be asked to provide the examiner with some form of positive identification.

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If the intent of a question is unclear, ask questions of the NRC examiner or the designated facility instructor only.

8. 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.

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

10. [illegible] you have turned in your examination, leave the examination area as defined by the proctor or NRC examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

11. Do you have any questions?

QUESTION 1 POINT VALUE: 1.00

Unit One (1) is operating with the following plant conditions:

Core Thermal Power	2558 MWth
Reactor pressure	1030 psig
Core Flow	77 Mlbm/hr

The SAFETY LIMIT for THERMAL POWER is the MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than:

a. 1.07

b. 1.08

c. 1.09

d. 1.10

QUESTION 2 POINT VALUE: 1.00

A Unit Two (2) startup is in progress per GP-02. The Reactor is critical with Reactor power at the point of adding heat.

Coolant temperature is being raised to saturation conditions, with Reactor steam dome pressure at 0 psig. RWCU is in service.

2PT-01.7, Heatup/Cooldown Monitoring is being performed. Which of the following conditions would result in **UNSATISFACTORY** Acceptance Criteria of the PT?

a. G31-TI-R607, Pt 5 indicates 169°F
C12-TR-R018, Ch 151 indicates 164°F
Reactor water level is 200"

b. G31-TI-R607, Pt 5 indicates 187°F
C12-TR-R018, Ch 151 indicates 183°F
Reactor water level is 210"

c. G31-TI-R607, Pt 5 indicates 174°F
C12-TR-R018, Ch 151 indicates 170°F
Reactor water level is 200"

d. G31-TI-R607, Pt 5 indicates 198°F
C12-TR-R018, Ch 151 indicates 195°F
Reactor water level is 210"

QUESTION 3 POINT VALUE: 1.00

Unit One (1) is operating with the following conditions:

Reactor Power	55%
Reactor Feed Pump 1A	operating
Reactor Feed Pump 1B	idling
Recirculation Pump Speeds	58%

Reactor Feed Pump (RFP) 1A trips and Reactor Level Hi/Lo alarms. Reactor Level drops to the scram setpoint and continues to lower to +110 inches before the operator brings the idling RFP on line (30 seconds after the trip of RFP 1A) to restore Reactor Level.

What should be the present status of the Recirculation Pumps?

a. Running at 58% speed.

b. Running on limiter #2.

c. Running on limiter #1.

d. Tripped on ATWS ARI/RPT.

QUESTION 4 POINT VALUE: 1.00

Unit Two (2) power ascension is in progress per GP-04. The operator is performing OP-02, Section 5.3, Speed/Power Increase Using the Recirculation Pump A(B) Speed Control (an Information Use procedure).

While raising the Recirculation Pumps' speed, the operator is cautioned to maintain Recirculation Pumps A and B within:

a. 5% when below 58 Mlbm/hr core flow, and 10% when ≥58 Mlbm/hr core flow.

b. 10% when below 58 Mlbm/hr core flow, and 5% when ≥58 Mlbm/hr core flow.

c. 10% when below 58 Mlbm/hr core flow, and 20% when ≥58 Mlbm/hr core flow.

d. 20% when below 58 Mlbm/hr core flow, and 10% when ≥58 Mlbm/hr core flow.

QUESTION 5 POINT VALUE: 1.00

Unit Two (2) has inserted a manual Reactor scram due to lowering condenser vacuum. Control rods failed to insert on the scram. Plant conditions:

Reactor power	31%
Steam flow	3.2 Mlbm/hr
Reactor pressure	960 psig, controlled by EHC
Drywell pressure	0.6 psig
Mode Switch	RUN
Main Turbine	Tripped on low vacuum

The operator is performing LEP-02, Section 3 to reset and scram the Reactor. Jumpers to bypass RPS trip signals have been requested but **NOT** yet installed. Which of the following would prevent the operator from resetting RPS prior to jumper installation?

a. Scram discharge volume Hi Hi level RPS trip sealed in.

b. Turbine stop valves closed with reactor power above 30%.

c. Reactor water level is controlling at the setdown setpoint.

d. IRMs upscale Hi Hi due to being inserted but not ranged up.

QUESTION 6 POINT VALUE: 1.00

Unit One (1) is operating at 27% power during Unit startup. The Turbine Generator has been synchronized to the grid. A total loss of Division II DC Switchboard 1B results in a reactor scram. During this transient:

57 control rods fail to fully insert
Reactor pressure peaks at 1132 psig
Reactor water level lowers to +107 inches
BOP Buses fail to transfer to the SAT
Diesel Generator 2 fails to start

What is the expected status of the Alternate Rod Injection (ARI) system? ARI has:

a. auto initiated on high reactor pressure.

b. auto initiated on low reactor water level.

c. not auto initiated but can be manually initiated.

d. not auto initiated and cannot be manually initiated.

QUESTION 7 POINT VALUE: 1.00

Following a line break on Unit Two (2) plant conditions are:

Reactor pressure	450 psig
Reactor water level	+70 inches
Drywell pressure	4.8 psig
Drywell temp (average)	165 deg F

All Drywell Cooler Fans will:

a. trip but can be restarted per SEP-10.

b. trip and cannot be restarted per SEP-10.

c. auto start but can be tripped at the RTGB.

d. auto start and cannot be tripped at the RTGB.

QUESTION 8 POINT VALUE: 1.00

Unit Two (2) is in OPERATIONAL CONDITION 5, with CORE ALTERATIONS in progress and SECONDARY CONTAINMENT INTEGRITY established.

The Interruptible Instrument Air Header ruptures and the Interruptible Air Header isolation valves are closed. Non-Interruptible Air Header pressure is normal.

How is SECONDARY CONTAINMENT INTEGRITY affected?

a. Reactor Building Supply and Exhaust Fans remain in service, SBGT auto starts.

b. Reactor Building Supply and Exhaust Fans remain in service, SBGT remains in standby.

c. Reactor Building Supply and Exhaust Fans trip and SBGT auto starts to maintain negative pressure.

d. Reactor Building Supply and Exhaust Fans trip but SBGT must be manually started to maintain negative pressure.

QUESTION 9 POINT VALUE: 1.00

Unit Two (2) is operating at power with Diesel Geneator 3 (DG3) under clearance. A lockout of a BOP Bus initiates a transient resulting in a reactor scram signal and an ATWS. Plant conditions:

Reactor power	10%
Bus E3	De-energized
SLC Switch	PUMP B RUN position
Bus E7-E8	Cross-tie Breakers racked in

How will the SLC system respond when the Bus E7-E8 Cross-tie Breakers are closed? SLC squib valve:

a. A fires, no SLC pump starts.

b. B fires, no SLC pump starts.

c. A fires, one SLC pump starts.

d. B fires, one SLC pump starts.

QUESTION 10 POINT VALUE: 1.00

The unit is at 10% power during reactor startup. The operator withdraws rod 22-19 to position 48. The following indications are noted:

Rod Drift alarm seals in
Rod Overtravel alarm seals in
Rod 22-19 four rod display is blank

What operator action is required?

a. Enter substitute rod position data into the RWM.

b. Insert rod 22-19 to position 46 to attempt recoupling.

c. Insert 22-19 to a position with an operable reed switch.

d. Fully insert rod 22-19 and disarm the HCU electrically or hydraulically.

QUESTION 11 POINT VALUE: 1.00

A Unit One (1) reactor scram occurs from 100% power. The operator completes the immediate scram actions and notes the following Rod Worth Minimizer (RWM) display:

ALL RODS IN:	NO
SHUTDOWN:	YES
RODS NOT FULL IN:	001

When the operator depresses the List Rods RWM key, rod 02-51 is displayed. What can be the furthest withdrawn position of rod 02-51 to cause the above display, and the basis for that rod position?

a. 48, shutdown margin calculations.

b. 06, maximum subcritical bank withdrawal position.

c. 04, maximum subcritical bank withdrawal position.

d. 02, maximum subcritical bank withdrawal position.

QUESTION 12 POINT VALUE: 1.00

Which of the following valves must be closed during performance of PT-14.2.1, Rod Scram Time Testing, to avoid erroneous scram times?

a. V105, exhast header isolation valve.

b. V113, charging header isolation valve.

c. V103, drive water header isolation valve.

d. V104, cooling water header isolation valve.

QUESTION 13 POINT VALUE: 1.00

During a low water level condition, CRD Flow maximization is being implemented per SEP-09 with the Reactor Building accessible.

The operator is directed to maintain Charging Water Header pressure ≥950 psig while opening the Flow Control and Pressure Control valves.

This limitation will prevent pump:

a. trip on overcurrent protection.

b. trip due to low suction pressure.

c. flowrate in excess of runout capacity.

d. discharge pressure dropping below reactor pressure.

QUESTION 14 POINT VALUE: 1.00

The initial reactor startup is in progress following refueling per GP-02. Initial criticality has just been achieved and PT-14.3.1, Insequence Critical Shutdown Margin Calculation, is being performed.

The operator notes the following SRM readings:

SRM Channel A	8.0 E4
SRM Channel B	7.0 E4
SRM Channel C	1.0 E5
SRM Channel D	5.6 E5

All IRMs are on Range 3. What automatic protective functions (if any) should have occurred?

a. Alarm only.

b. Rod block only.

c. Rod block and 1/2 scram.

d. Rod block and full scram.

QUESTION 15 POINT VALUE: 1.00

During a Reactor startup following refueling, the operator is performing PT-50.2, IRM Range 6/7 Overlap Determination. The following data is recorded:

	Range 6	Range 7
IRM A	50	4
IRM B	42	5
IRM C	45	5
IRM D	43	4
IRM E	37	3
IRM F	52	6
IRM G	44	4
IRM H	50	6

The Level 2 acceptance criteria of PT-50.2 is:

a. satisfactory for all Division I and II IRMs.

b. unsatisfactory for at least one Division I IRM only.

c. unsatisfactory for at least one Divivision II IRM only.

d. unsatisfactory for at least one Division I and one Division II IRM.

QUESTION 16 POINT VALUE: 1.00

APRM Channel C has all associated LPRM inputs Operable, with the LPRM function switches in Operate. During performance of a MST, the LPRM function switches are placed to and left in Bypass, one at a time.

What is the MINIMUM number of LPRM function switches that should have been placed in Bypass when a Neutron Monitoring trip signal (1/2 scram) is received?

a. 3

b. 4

c. 6

d. 7

QUESTION 17 POINT VALUE: 1.00

Unit Two (2) is operating at 55% power. Control Rod withdrawal is in progress for plant startup. The following indications and alarms occur:

Rod 46-11 is selected
Rod Out Block alarm sealed in
APRM Downscale/Inop alarm sealed in
APRM Channel A indicates 0% power

Rod Block Monitor (RBM) Channel A will:

a. be bypassed automatically.

b. remain in normal operation enforcing rod blocks.

c. fail downscale resulting in a rod block.

d. transfer to APRM Channel E for reference power.

QUESTION 18 POINT VALUE: 1.00

Unit Two (2) is operating at 100% power. The Standby Gas Treatment System (SBGT) is in the standby alignment. A Trip of both Reactor Feedwater Pumps results in the following plant conditions:

Reactor water level	+80 inches
Reactor pressure	945 psig
Drywell pressure	+0.3 psig
Rx Bldg pressure	-0.4" WC

HPCI and RCIC have initiated to restore Reactor water level. All systems respond as designed during the transient. What operator action is <u>required</u> concerning the SBGT System?

a. Secure one SBGT fan per OP-10.

b. Open Post LOCA Vent valves (SGT-V8 and V9).

c. Open Primary Containment Suction valve (VA-2F-BFV-RB).

d. Restart Rx Bldg HVAC and secure both SBGT trains per SEP-04.

QUESTION 19 POINT VALUE: 1.00

Following a LOCA, the Containment Hydrogen/Oxygen monitors have been placed in service using CAM overrides per the Hard Card.

After placing the monitors in service, the following alarms and indications occur:

Reactor Building Vent Rad Hi alarm sealed in
Reactor Building Vent Rad Hi Hi alarm sealed in
Reactor Building Vent Radiation recorder channel A pegged high
Reactor Building Vent Radiation recorder channel B 0.5 mr/hr

How does this failure affect Hydrogen/Oxygen (H2/O2) monitors?

a. Both Division I and II H2/O2 monitors remain in service.

b. Only Division I H2/O2 monitor isolates, but can be placed back in service.

c. Only Division I H2/O2 monitor isolates, and cannot be placed back in service.

d. Both Division I and II H2/O2 monitors isolate, but can be placed back in service.

QUESTION 20 POINT VALUE: 1.00

Unit One (1) is operating at rated power, when the loss of an electrical power distribution system results in the following Group 1 PCIS status light indications on P601:

Inboard MSIV DC solenoid	Out
Inboard MSIV AC solenoid	Out
Outboard MSIV DC solenoid	Out
Outboard MSIV AC solenoid	Lit

What power distribution system has been lost?

a. Division I AC.

b. Division I DC.

c. Division II AC.

d. Division II DC.

QUESTION 21 POINT VALUE: 1.00

During severe accident conditions, E&RC has been directed to take samples at the Post Accident Sample Station to determine the extent of core damage.

They have requested that the control room open the RHR Heat Exchanger sample valves (E11-F079/F080). The valid Group 2 isolation signal for these valves is overriden by placing:

a. CAM Div I/Div II Isol Ovrd switches on XU51 to ON.

b. Heat exchanger sample control on XU-75/79 to LOCAL.

c. CAC Div I/Div II AC Isol Ovrd switches on XU51 to ON.

d. Control Power For PASS Isol Valves switches on XU75/XU79 to ON.

QUESTION 22 POINT VALUE: 1.00

Unit Two (2) is operating at power when BOP Bus 2C trips on bus lockout. The associated Diesel Generator auto starts and energizes its E Bus. One minute later, a Reactor scram occurs. Following the scram:

117 control rods fail to fully insert
SLC system is intiated with both SLC pumps
Reactor water level is lowered to +125 inches, after SLC is started

The operator notes that the RWCU system Inboard Isolation Valve (F001) and the Outboard Isolation Valve (F004) are both closed. What closed the valves?

a. F001 and F004 both closed when power was lost.

b. F001 closed on low RPV level, F004 closed when power was lost.

c. F001 closed on low RPV level, F004 closed when SLC was started.

d. F001 closed when power was lost, F004 closed when SLC was started.

QUESTION 23 POINT VALUE: 1.00

Unit One (1) is performing Alternate Emergency Depressurization. RCIC has been placed into pressure control to aid in Reactor pressure reduction.

Circuit alterations have been performed per EOP-01-RVCP and EOP-01-SEP-10 for HPCI and RCIC.

Which of the following conditions, by interlock, will remove the RCIC System from the pressure control mode of operation?

a. Reactor pressure lowers to 50 psig.

b. Drywell pressure rises to 2.5 psig.

c. Reactor water level lowers to +110 inches.

d. Suppression pool level rises to -23 inches.

QUESTION 24 POINT VALUE: 1.00

Unit One (1) failed to scram. RHR is placed in Suppression Pool Cooling per the Hard Card without use of overrides. Reactor water level is then deliberately lowered to suppress power. Plant conditions are:

Reactor water level	-45 inches (N036/N037)
Reactor pressure	800 psig
Drywell pressure	0.6 psig
Suppression pool temp	130°F

Suppression Pool Cooling valves have closed. Returning Suppression Pool Cooling to service requires:

a. placing the Think Switch to Manual only.

b. bypassing the 2/3 core height interlock only.

c. placing the Think Switch to Manual, then bypassing the 2/3 core height interlock.

d. bypassing the 2/3 core height interlock, then placing the Think Switch to Manual.

QUESTION 25 POINT VALUE: 1.00

Consider the following normal RHR suction valve interlocks for control of Reactor water level:

1. Shutdown Cooling suction isolation valves (F008/F009) will automatically isolate on low Reactor water level.
2. Shutdown Cooling pump suction valve (F006) cannot be opened unless Torus common suction valve (F020) is closed.

During Shutdown Cooling operation from Remote Shutdown stations per AOP-32.0, which of the above (if any) interlocks are functional?

a. 1 only.

b. 2 only.

c. both 1 and 2.

d. neither 1 nor 2.

QUESTION 26 POINT VALUE: 1.00

Unit One (1) is operating at rated power when a Loss Of Off-site Power occurs simultaneously with a line break in the drywell. Plant conditions are:

Reactor water level	-50" (N036/N037)
Reactor pressure	150 psig
Drywell pressure	20 psig
Diesel Generator 2	Tripped

No E Bus cross-tie actions have been performed. What is the expected status of RHR Loop 1B?

a. Pump 1B running under dead head conditions.

b. Pump 1D running under dead head conditions.

c. Pump 1B running with injection to the reactor.

d. Pump 1D running with injection to the reactor.

QUESTION 27 POINT VALUE: 1.00

Following a small line break, RHR Loop A is placed in Drywell and Suppression Chamber Spray using required overrides per SEP-02 and SEP-03. Plant conditions are:

Reactor Water Level	+175 inches
Reactor Pressure	900 psig
Drywell Pressure	15.0 psig

Reactor water level drops to -60 inches. How will RHR Loop A and RHR Service Water (RHR SW) Loop A respond?

The RHR Loop A drywell/suppression chamber spray valves:

a. auto close, RHR SW Loop A pump(s) trip.

b. auto close, RHR SW Loop A remains running.

c. remain open, RHR SW Loop A pump(s) trip.

d. remain open, RHR SW Loop A remains running.

QUESTION 28 POINT VALUE: 1.00

During normal power operation of Unit Two (2), an ECCS Division I Trip Cabinet Trouble alarm is received. Investigation shows that <u>BOTH</u> power supplies to the trip cabinet (XU-63) have failed and all associated trip unit meters indicate downscale with no trip lights lit.

A DBA LOCA then occurs resulting in Reactor water level rapidly dropping below the Top of Active Fuel and rapid Reactor depressuriztion. How will Division I Low Pressure ECCS (Core Spray 2A and RHR LPCI Loop 2A) respond?

a. Core Spray 2A initiates, LPCI 2A fails to initiate.

b. Core Spray 2A and LPCI 2A both fail to initiate.

c. Core Spray 2A fails to initiate, LPCI 2A initiates.

d. Core Spray 2A and LPCI 2A will both auto initiate.

QUESTION 29 POINT VALUE: 1.00

Following a large line break on Unit Two (2), Core Spray Pump 2A is injecting to the reactor. Core Spray Pump 2B has been overriden off. Plant conditions are:

Reactor water level	200"
Reactor pressure	25 psig
Suppression pool level	-43 inches
Suppression pool temperature	210°F.
Suppression chamber pressure	5.5 psig.

Core Spray Pump 2A flow rate is currently 5500 gpm. The operator should reduce Core Spray Pump 2A flow rate by throttling:

a. Inboard Injection valve (E21-F005A) to approximately 3000 gpm.

b. Inboard Injection valve (E21-F005A) to approximately 4500 gpm.

c. Outboard Injection valve (E21-F004A) to approximately 3000 gpm.

d. Outboard Injection valve (E21-F004A) to approximately 4500 gpm.

QUESTION 30 POINT VALUE: 1.00

Unit Two (2) HPCI has automatically initiated on a valid initiation signal. The operator observes the following indications:

Steam Supply Pressure	0 psig
Turbine Exhaust Pressure	0 psig
Pump Discharge Pressure	0 psig
Turbine Speed	1000 RPM, lowering

HPCI TURB TRIP SOL ENERG <u>**NOT**</u> Alarming

Which of the following would explain the above indications?

a. Isolation due to ruptured exhaust diaphram.

b. Overspeed trip from speed feedback signal failure.

c. Loss of oil pressure to the turbine control system.

d. Loss of 125 VDC input to the 24/52.5 VDC power supplies.

QUESTION 31 POINT VALUE: 1.00

A Reactor scram has occurred due to a Group 1 isolation. ADS/SRVs are required for Reactor pressure control.

During verification of Group Isolations, the operator inadvertantly closed the PNS supply to the drywell isolation valves, and failed to notice the Backup Nitrogen valves were closed.

No Group 10 isolation signal is received during the transient. How will the operation of ADS/SRVs be affected?

a. Only ADS valves can be opened manually until the accumulator supply is depleted.

b. Both ADS and SRV valves can be opened manually until the accumulator supply is depleted.

c. Only ADS valves will be supplied by Backup Nitrogen when drywell pneumatic header drops below 95 psig.

d. Both ADS and SRV valves will be supplied by Backup Nitrogen when drywell pneumatic header drops below 95 psig.

QUESTION 32 POINT VALUE: 1.00

Unit One (1) has experienced a high Reactor pressure transient following a Main Turbine trip. Current plant conditions:

All rods in
Reactor Pressure 950 psig controlled by EHC
Eleven (11) SRV's green indicating lights lit
Eight (8) amber memory lights illuminated

Determine the extent of the pressure transient.

a. 1122 psig

b. 1132 psig

c. 1142 psig

d. 1152 psig

QUESTION 33 POINT VALUE: 1.00

Unit One (1) is operating at power with Core Spray Pump 1B under clearance. A Loss of Off-site Power occurs to BOTH units. Diesel Generators (DGs) 2 and 4 auto start, DGs 1 and 3 trip and lockout.

A stuck open SRV and loss of high pressure injection causes Reactor water level to lower. All available low pressure ECCS pumps have been manually started. Plant conditions are:

Reactor water level	LPCI/Core Spray initiation signal just received
Reactor pressure	650 psig
Drywell pressure	0.6 psig
ADS Inhibit Switches	AUTO

Assuming **NO** operator action, ADS will:

a. Auto initiate in 1 minute 23 seconds.

b. Auto initiate in 1 minute 45 seconds.

c. Not auto initiate due to lack of high drywell pressure.

d. Not auto initiate due to lack of ECCS pump permissive.

QUESTION 34 POINT VALUE: 1.00

Unit Two (2) has a Loss Of Off-Site Power. HPCI and RCIC both failed. Reactor Water Level dropped to +30 inches before CRD reversed the level trend. Current conditions are:

Reactor water level	+60 inches, rising
Reactor pressure	800-1000 psig using SRVs
Drywell pressure	2.2 psig, rising
Average drywell temp	180°F, rising
Generator primary lockout	Tripped

RBCCW Pumps are tripped, and Nuclear Service Water (NSW) cooling water valves (SW-V103/V106) are closed. What actions are required to restore RBCCW to control containment parameters?

a. Reset the Primary Generator lockout, align RBCCW cooling to the conventional header.

b. Reset the Primary Generator lockout, reopen NSW cooling water valves SW-V103/V106.

c. Reset Core Spray initiation logic, align RBCCW cooling to the conventional header.

d. Reset Core Spray initiation logic, reopen NSW cooling water valves SW-V103/V106.

QUESTION 35 POINT VALUE: 1.00

During full power operation, an unisolable Instrument Air Header rupture occurs. The Non-Interruptible Instrument Air header isolation valves (IAN-V50 and IAN-V51) are closed per the AOP. The operator notes the following:

BOTH Reactor Building Standby Air Compressors fail to start.
Backup Nitogen automatically aligns on low RNA header pressure.
PNS header pressure remains normal.

Assume no Group 1 isolation signals. How will the MSIVs respond to the loss of pneumatics?

a. Inboard and Outboard MSIVs remain open.

b. Inboard and Outboard MSIVs drift closed.

c. Inboard MSIVs drift closed, Outboard MSIVs remain open.

d. Inboard MSIVs remain open, Outboard MSIVs drift closed.

QUESTION 36 POINT VALUE: 1.00

Following a Reactor scram, the operator trips the main turbine and notes the following turbine valve positions:

Turbine stop valves are closed
Turbine control valves are closed
Combined intermediate intercept and stop valves are open
One turbine bypass valve open 25%

This condition could result in turbine damage caused by:

a. overspeeding of the turbine.

b. excessive axial thrust on the shaft.

c. overpressurization of the LP shells.

d. excessive moisture impingement on LP blades.

QUESTION 37 POINT VALUE: 1.00

Unit Two (2) is operating steady state at rated power with the following Electro Hydraulic Control (EHC) conditions:

Reactor pressure	1005 psig
EHC Pressure Setpoint	920 psig
PAM pressure	950 psig
Pressure regulator A	In control
Pressure regulator B	5 psig biac

The Pressure Averaging Manifold (PAM) pressure input to Pressure Regulator A fails low. The PAM pressure input to Pressure Regulator B is unaffected. How will the EHC System respond?

a. Pressure regulator B takes control and stabilizes PAM pressure at 945 psig.

b. Pressure regulator B takes control and stabilizes PAM pressure at 955 psig.

c. Control valves close, reactor pressure and neutron flux rise and the reactor scrams.

d. Control/Bypass valves open and steam line pressure lowers to the Group 1 isolation setpoint.

QUESTION 38 POINT VALUE: 1.00

Unit One (1) is operating at 100% power with the following Condensate system alignment:

Condensate pumps	A & B Running
Condensate pump	C Standby/Auto
Condensate Booster pumps	B & C Running
Condensate Booster pump	A Under clearance

4KV BOP Bus 1D trips and locks out due to a bus fault. The transient results in a Reactor scram. What is the availability of the Condensate system to provide makeup to the Reactor vessel?

Condensate pump:

a. A is available, no Condensate Booster pump is available.

b. B is available, no Condensate Booster pump is available.

c. A is available and Condensate Booster pump C is available.

d. B is available and Condensate Booster pump C is available.

QUESTION 39 POINT VALUE: 1.00

Unit Two (2) is operating at 100% power. The Digital Feedwater Control System (DFCS) is aligned as follows:

Master Controller	Auto, set at 187 inches
Level instrument N004A	187 inches
Level instrument N004B	187 inches
Level instrument N004C	failed downscale
Mode Select switch	3 Element
Level Select switch	Level A

Level instrument N004A fails downscale. Assuming no operator action, Reactor water level will:

a. rise and flood the main steam lines.

b. drop to the low level scram setpoint.

c. rise resulting in a main turbine and feed pump trip.

d. remain at 187" with level instrument N004B in control.

QUESTION 40 POINT VALUE: 1.00

Unit Two (2) is in operation at 100% power. The following annunciator and conditions exist:

A-07, 1-2 RFP FW CONTROL SIGNAL FAILURE
Amber light above A RFPT lockout switch out
Amber light above B RFPT lockout switch lit

A Reactor Recirculation Pump trips. Reactor power lowers and Reactor water level begins to rise uncontrollably. How can the operator restore Reactor water level to the normal band?

a. Operate RFP A MSC control in the Lower direction.

b. Operate RFP B MSC control in the Lower direction.

c. Place RFP A MGU in Manual and lower output demand.

d. Place RFP B MGU in Manual and lower output demand.

QUESTION 41 POINT VALUE: 1.00

The Control Building Ventilation system has initiated in the radiation protection mode. The alignment of system controls is:

Emergency Filtration Fan A	Pref
Emergency Filtration Fan B	Stby

Emergency Filtration Fan A has been running for 10 minutes when a high temperature is sensed in the Train A charcoal bed. Assuming the radiation initiation signal is still present, Emergency Filtration Fan A will:

a. immediately trip and Fan B will immediately auto start.

b. remain running since the high temperature trip is bypassed.

c. immediately trip and Fan B will start after a 10 second delay.

d. remain running since the high temperature provides alarm only.

QUESTION 42 POINT VALUE: 1.00

A Loss of Off-Site Power has occured. Secondary Containment isolated. Reactor Building Ventilation was restarted by guidance of EOP-03-SCCP using SEP-04. Plant conditions are:

Reactor Water Level is +150 inches, slowly rising
Drywell Pressure is 1.2 psig, slowly rising
CAC Vent Purge Isol Ovrd (CAC-CS-5519) is in OVERRIDE
Reactor Building Vent Rad Monitors have been reset
PCIS Isolation Reset push buttons on P601 have been depressed

Which of the following would cause the Reactor Building to re-isolate?

a. Drywell pressure rises above 2.0 psig.

b. Reactor level drops to the Top Of Active Fuel.

c. Main Stack Radiation Monitor exceeds the Hi-Hi setpoint.

d. Reactor Building Vent Exhaust temperature exceeds 140°F.

QUESTION 43 POINT VALUE: 1.00

Following a Loss of Off-Site Power, Diesel Generator #1 (DG1) is running in AUTO, tied to Bus E1. DG1 parameters:

Kilowatt load	3500 KW
Terminal Voltage	4160 Volts
Reactive load	1300 KVAR
Frequency	60 Hz

Off-Site Power has been restored and the BOP buses energized from the SAT. The BOP bus to E1 Master/Slave breaker is still open.

The operator depresses the DG #1 CONTROL ROOM MANUAL push button on the RTGB. DG #1 frequency will be approximately:

a. 57 Hz.

b. 59 Hz.

c. 61 Hz.

d. 63 Hz.

QUESTION 44 POINT VALUE: 1.00

During performance of PT-12.2C, Diesel Generator (DG) 3 has TRIPPED on reverse power due to misoperation of the governor control switch. A Loss of Off-Site Power then occurs.

What operator action is required to start DG3 following the Loss of Off-Site Power?

a. Reset the engine lockout only.

b. Reset the generator lockout only.

c. Reset the engine lockout, then the generator lockout.

d. Reset the generator lockout, then the engine lockout.

QUESTION 45 POINT VALUE: 1.00

The following sequence of events occurs on Unit One (1):

Time = 0 seconds	Off-site power is lost
Time = 5 seconds	A LOCA signal is recieved
Time = 10 seconds	Diesel generators energize thier respective E Buses

The Motor Driven Fire Pump normal feeder breaker from:

a. Bus E1 closes at Time = 25 seconds.

b. Bus E2 closes at Time = 25 seconds.

c. Bus E1 closes at Time = 30 seconds.

d. Bus E2 closes at Time = 30 seconds.

QUESTION 46 POINT VALUE: 1.00

Unit One (1) is in an outage with the SAT energized and the UAT in backfeed alignment. All backfeed selector switches are in the BACKFEED position. Electrical system alignment:

BOP Buses 1C/1D	Powered from UAT
E Buses E1/E2	Powered from BOP Buses
DGs 1, 2, 3, 4	Operable in standby alignment

A sudden fault pressure occurs in the Main Power Transformer resulting in a Backup Main Generator lockout. How will the electrical distribution system respond?

a. BOP buses 1C/1D transfer to the SAT, four DGs receive an auto start signal.

b. BOP buses 1C/1D transfer to the SAT, no DGs receive an auto start signal.

c. BOP buses 1C/1D are de-energized, four DGs receive an auto start signal.

d. BOP buses 1C/1D are de-energized, DGs 1 and 2 only receive an auto start signal.

QUESTION 47 POINT VALUE: 1.00

A fire in Diesel Generator (DG) Cell #1 has required entry into ASSD-03 for safe shutdown of <u>both</u> units utilizing Safe Shutdown Train B. Status of the electrical plant is:

DG 3 and 4	Running, tied to associated E Buses
DG 1 and 2	Tripped, not available
4KV Bus E2	Energized from 4KV Bus E1 via E3
E3 to E1 to E2	Cross-tie breakers in FIRE

What will trip the E1 to E2 cross-tie breakers?

a. LOCA signal on Unit 1 or Unit 2.

b. DG3 high lube oil temperature.

c. DG3 ground overcurrent.

d. DG3 differential overcurrent.

QUESTION 48 POINT VALUE: 1.00

A "250 V BATT A GROUND" alarm has been received on Unit One (1). The following readings are reported from the Battery Room:

P Bus	0.6 milliamps
N Bus	3.4 milliamps
Charger 1A-1	135 volts, in float
Charger 1A-2	135 volts, in float

Per OP-51 and AI-115, the ground is on the:

a. P Bus, action level 1 applies.

b. P Bus, action level 2 applies.

c. N Bus, action level 1 applies.

d. N Bus, action level 2 applies.

QUESTION 49 POINT VALUE: 1.00

Unit One (1) is operating at 100% power. The UPS system is in its normal alignment for <u>both</u> units. A total loss of UPS occurs on Unit One (1), followed shortly by a spurious Reactor scram. Plant conditions:

Reactor water level lowers to +135 inches
Reactor pressure 950 psig controlled by EHC
APRM recorders on P603 indicate 100% power
Digital Feedwater controller displays are blank

How will the loss of UPS affect the plant during this transient?

a. Reactor power cannot be determined to be less than 3% from P603.

b. Reactor feed pumps will not respond to the reduced Reactor water level.

c. EHC pressure control will be lost as the main turbine coasts down to zero speed.

d. Reactor Building HVAC is lost until the main stack rad monitor is transferred to Unit Two.

QUESTION 50 POINT VALUE: 1.00

An Auxiliary Operator has received 1.95 rem TEDE for the current year. The AO is needed to perform work in a 25 mrem/hr field. The work is expected to last 3 hours.

In accordance with NGGM-PM-0002, Radiation Control and Protection Manual, the worker requires:

a. approval from the Manager - E&RC to exceed the annual administrative dose limit.

b. approval from the Plant General Manager to exceed the annual administrative dose limit.

c. approval from the Site Vice President to exceed the annual administrative dose limit.

d. no special authorizations since the annual administrative limit should not be exceeded.

QUESTION 51 POINT VALUE: 1.00

Both Units have lost off-site power. The only available Diesel Generator is DG2. Buses E2 and E4 cannot be cross-tied. Unit Two (2) UPS has been de-energized for DC load stripping.

What instruments are available to monitor Reactor Water Level on the Unit Two (2) RTGB?

a. Fuel Zone indicator N036 only.

b. Fuel Zone indicator N037 only.

c. Fuel Zone indicator N036 and Narrow Range indicators N004A/B/C.

d. Fuel Zone indicator N037 and Narrow Range indicators N004A/B/C.

QUESTION 52 POINT VALUE: 1.00

You are escorting a visitor with a red badge in the protected area. It becomes desired to temporarily give up your escort duties.

How may this be accomplished?

a. You and the visitor must exit the protected area.

b. You may turn over escort duties to a security guard only.

c. You may turn over escort duties to any other qualified escort, and notify security at the access point you entered.

d. You may turn over escort duties to any other qualified escort, and notify security at the secondary alarm station.

QUESTION 53 POINT VALUE: 1.00

A Temporary Procedure Change is developed and designated as Revision To Follow. This Temporary Change receives:

Interim approval on March 2nd
Final approval on March 6th

What is the last date this Temporary change may be used <u>WITHOUT</u> receiving any allowable extention(s)?

a. May 1st.

b. May 5th.

c. May 31st.

d. June 4th.

QUESTION 54 POINT VALUE: 1.00

Which of the following describes the level of use of Emergency Operating Procedures?

a. Reference use procedures.

b. Continuous use procedures.

c. Information use procedures.

d. Exempt from level of use requirements.

QUESTION 55 POINT VALUE: 1.00

Per PLP-21, which of the following would be an UNACCEPTABLE method of performing independent verification by use of RTGB indications?

Independent verification of a Core Spray System:

a. pump breaker closure using red light indication.

b. pump breaker rack in status using green light indication.

c. injection valve opening using system flow indication meter.

d. injection valve standby position using green light indication.

QUESTION 56 POINT VALUE: 1.00

A RCIC valve lineup checksheet is being performed near the end of a refueling outage. The working copy has been verified in accordance with the requirements of AP-008.

Per AP-008, the working copy is <u>REQUIRED</u> to be reverified:

a. daily using a controlled copy, or every 3 days using NRCS.

b. daily using a controlled copy, or every 3 days by contacting document services.

c. every 3 days using a controlled copy, or every 7 days using NRCS.

d. every 3 days using a controlled copy, or every 7 days by contacting document services.

QUESTION 57 POINT VALUE: 1.00

During RTGB walkdown in preparation for shift turnover, the oncoming Reactor Operator notes that annunciator A-5 2-2, Rod Out Block:

Alarm is sealed in
Has a yellow dot affixed to the window

If the signal causing this alarm clears, then comes back in without the alarm being reset, the subsequent alarm condition is indicated by the alarm window flashing:

a. slowly with an audible alarm.

b. rapidly with an audible alarm.

c. slowly, then rapidly without an audible alarm.

d. rapidly, then slowly without an audible alarm.

QUESTION 58 POINT VALUE: 1.00

A motor-operated valve has been manually backseated. This valve is:

non safety related
normally operated from the RTGB

AP-013 requires that the valve:

a. control switch is caution tagged denoting the valve is backseated.

b. handwheel is locked in position by use of an approved valve locking device.

c. must be manually operated off of the backseat prior to motor operation.

d. motor breaker is placed under clearance in the off position.

QUESTION 59 POINT VALUE: 1.00

A plant shutdown is in progress per GP-05 for refueling. Current plant conditions are:

Reactor power 20%
Drywell oxygen 19.0%

Drywell entry may be made with the approval of the Manager-E&RC and the authorization of the Shift Superintendent only if reactor power is reduced at least:

a. 5%, oxygen concentration is acceptable.

b. 5%, oxygen concentration must be raised.

c. 15%, oxygen concentration is acceptable.

d. 15%, oxygen concentration must be raised.

QUESTION 60 POINT VALUE: 1.00

Which of the following valve/actuator types may be used as a clearance boundary isolation component, provided the associated restrictions of AI-58 are satisfied?

a. Solenoid operated ball valve marked as fail-closed on the print.

b. Motor operated globe valve normally used as a flow control valve.

c. Pressure balanced diaphragm operated pinch valve not equipped with a handwheel.

d. Double acting cylinder operated butterfly valve not marked as fail-closed on the print.

QUESTION 61 POINT VALUE: 1.00

A clearance request has been received for a fluid system with the following normal operating parameters:

Pressure	475 psig
Temperature	175°F

Per AI-58, the clearance:

a. May use single valve boundary isolation.

b. Must use dual valve boundary isolation due to pressure only.

c. Must use dual valve boundary isolation due to temperature only.

d. Must use dual valve boundary isolation due to pressure and temperature.

QUESTION 62 POINT VALUE: 1.00

During accident conditions on Unit One (1), the following plant conditions exist:

Reactor water level	-70 inches (Fuel Zone)
Reactor pressure	1100 psig
Drywell average temp	190°F
Drywell ref leg area temp	270°F
Injection sources	None available

Under these conditions, peak fuel clad temperature will not exceed:

a. 1500°F, provided Reactor water level remains above -80 inches.

b. 1500°F, provided Reactor water level remains above -90 inches.

c. 1800°F, provided Reactor water level remains above -80 inches.

d. 1800°F, provided Reactor water level remains above -90 inches.

QUESTION 63 POINT VALUE: 1.00

Following accident conditions, the crew is executing the Reactor Vessel Flooding Procedure, EOP-01-RXFP. Plant conditions are:

Control rods	Fully inserted
Reactor water level	Unknown

The operator is directed to control injection flow to the Reactor to maintain at least _____ SRV/ADS Valves open and Reactor pressure:

a. 4; above the Minimum Alternate Flooding Pressure.

b. 5; above the Minimum Alternate Flooding Pressure.

c. 4; at least 50 psig above suppression chamber pressure.

d. 5; at least 50 psig above suppression chamber pressure.

QUESTION 64 POINT VALUE: 1.00

A heavy influx of marsh grass on the Circulating Water Screens has caused a loss of all Circulating Water pumps and a reactor scram. Plant conditions are:

Group 1 isolated
Condenser vacuum is 0" Hg
Turbine speed is 500 rpm, dropping
EHC Electrical Malfunction in alarm due to loss of the PMG.

The marsh grass is now cleared and the Circulating Water System has been restarted. Is the Main Condenser available as a heat sink?

a. No, the MSIVs are closed.

b. No, the EHC system is not available.

c. No, the condenser is not under vacuum.

d. Yes, all required systems are available.

QUESTION 65 POINT VALUE: 1.00

Following a loss of all high pressure injection on Unit One (1), seven ADS valves have been manually opened to restore adequate core cooling. Plant conditions are now:

Reactor water level	+25", N026A/B
Reactor water level	+140", N036/37
Reactor pressure	25 psig
Drywell average temp	155°F
Drywell ref leg area temp	280°F

Reactor water level may be determined using:

a. N026A/B only.

b. N036/37 only.

c. Both N026A/B and N036/37.

d. Neither N026A/B nor N036/37.

QUESTION 66 POINT VALUE: 1.00

A Unit Two (2) reactor scram has occurred. Seven control rods failed to fully insert and are between positions 08 and 18. Conditions are:

All APRM Downscale lights are LIT
MSIVs are open
Total Steam Flow 3.6 E6 lbm/Hr, dropping
Reactor Pressure 900 psig, dropping
Narrow Range Level Instruments (N004s) +155 inches, rising
Master Feedwater setpoint at +170"
Two Reactor Feed Pumps in operation

With current plant conditions, the operator is required as an IMMEDIATE action to:

a. trip the Main Turbine.

b. trip one Reactor Feed Pump.

c. place the Mode Switch to SHUTDOWN.

d. enter Alternate Control Rod Insertion.

QUESTION 67 POINT VALUE: 1.00

Following a group 1 isolation and a reactor scram, the operating crew is performing the Reactor Scram Procedure, EOP-01-RSP. Plant conditions are:

Reactor water level	195 inches, slowly rising (N004s)
Reactor pressure	800-1000 psig, controlled by SRVs
Drywell pressure	1.0 psig, slowly rising
Suppression pool temp	94°F, slowly rising
Suppression pool level	-27.5", slowly rising

The operating crew is required to enter EOP-01-RVCP and execute concurrently with the scram procedure if:

a. drywell pressure rises to 1.5 psig.

b. reactor water level rises to +230 inches.

c. suppression pool temperature rises to 111°F.

d. suppression pocl level rises to -26.5 inches.

QUESTION 68 POINT VALUE: 1.00

Following a Unit One (1) Reactor scram, the crew has entered and is executing EOP-01-RSP, Reactor Scram Procedure. Plant conditions are:

Reactor water level	220", rising
Reactor pressure	945 psig, stable
MSIVs	Open

Per EOP-01-RSP, the MSIVs must be manually closed if Reactor water level cannot be maintained below:

a. 230"

b. 240"

c. 250"

d. 260"

QUESTION 69 POINT VALUE: 1.00

The entry conditions for Unit One (1) EOP-01-RVCP, Reactor Vessel Control Procedure for Reactor pressure and water level are Reactor pressure is greater than:

a. 1035 psig, or Reactor water level less than 153".

b. 1035 psig, or Reactor water level less than 166".

c. 1060 psig, or Reactor water level less than 153".

d. 1060 psig, or Reactor water level less than 166".

QUESTION 70 POINT VALUE: 1.00

During ATWS conditions on Unit Two (2), injection to the Reactor has been terminated and prevented to suppress Reactor power. Plant conditions are:

Reactor water level	-35 inches (Fuel Zone), lowering
Reactor power	7%
Reactor pressure	950 psig
MSIVs	Closed
SRVs	One open
CAC-TR-4426-1B, Point A	195°F
CAC-TR-4426-1B, Point B	203°F
CAC-TR-4426-2B, Point A	187°F
CAC-TR-4426-2B, Point B	191°F

Assuming APRMs are NOT downscale, what is the HIGHEST indicated Reactor water level injection may be re-established to the Reactor?

a. -45 inches

b. -55 inches

c. -67 inches

d. -76 inches

QUESTION 71 POINT VALUE: 1.00

Following an incomplete Reactor scram, the operating crew is executing EOP-01-LPC, Level/Power Control. A decision step is reached asking "Is The Reactor Shutdown?".

Which of the following conditions would satisfy the definition of "SHUTDOWN" as it applies to the Reactor?

a. All operable APRMs indicate downscale.

b. The Reactor is subcritical on range 6 of IRMs.

c. The entire SLC Tank has been injected to the Reactor.

d. Hot Shutdown Boron Weight has been injected to the Reactor.

QUESTION 72 POINT VALUE: 1.00

Following a reactor scram and a group 1 isolation, SRVs are being used to maintain reactor pressure 900-1000 psig.

Which of the following conditions requires <u>ALL</u> group 1 isolations to be defeated and the reactor vessel rapidly depressurized to the main condenser?

a. Suppression Pool Level is +4' 6"
   Suppression Pool Temperature is 95°F

b. Suppression Pool Level is -1' 6"
   Suppression Pool Temperature is 170°F

c. Suppression Pool Level is -4' 3"
   Suppression Pool Temperature is 156°F

d. Suppression Pool Level is -8' 1"
   Suppression Pool Temperature is 105°F

QUESTION 73 POINT VALUE: 1.00

During accident conditions, Reactor Water level cannot be restored above the Top of Active Fuel. Service Water is injecting to the Reactor Vessel to raise Containment Level.

Service Water injection MUST be secured, IRRESPECTIVE of Adequate Core Cooling when Primary containment level reaches:

a. 63 feet, to prevent covering the highest reactor vessel vent path capable of rejecting all decay heat.

b. 63 feet, to prevent covering the highest primary containment vent path capable of rejecting all decay heat.

c. 68.5 feet, to prevent covering the highest reactor vessel vent path capable of rejecting all decay heat.

d. 68.5 feet, to prevent covering the highest primary containment vent path capable of rejecting all decay heat.

QUESTION 74 POINT VALUE: 1.00

Following a large Recirculation line rupture, EOP-01-PCFP, Primary Containment Flooding Procedure, is being exectuted. The following indications are available:

CAC-LI-2601-1	+5.9 feet
CAC-PI-1257-2A	23 psig
CAC-PI-1230	21 psig
CAC-PI-4176	25 psig
CAC-PR-1257-1	22 psig

What is Primary Containment water level?

a. +14.5 feet

b. +9.9 feet

c. +7.6 feet

d. +4.6 feet

QUESTION 75 POINT VALUE: 1.00

A seismic event has occurred that has resulted in a Loss of Off-Site Power and high power ATWS conditions.

The SLC Storage Tank outlet line completely severed at the tank during the earthquake. The SLC tank is <u>EMPTY</u> making the SLC pumps unavailable for boron injection.

Which system should be selected for alternate boron injection?

a. CRD

b. RCIC

c. RWCU

d. Condensate

QUESTION 76 POINT VALUE: 1.00

A Condensate header rupture in the cable spread area of the Control Building has resulted in a loss of all UPS and RPS power.

Plant status is as follows:

Blue scram lights	137 illuminated
IRM Indications	50 on Range 10

What method of EOP-01-LEP-02, Alternate Control Rod Insertion, would be MOST effective in inserting the withdrawn rods?

a. Vent the scram air header.

b. Vent the overpiston area of control rods.

c. Scram individual rods with the scram test switches.

d. Insert control rods with the Reactor Manual Control System.

QUESTION 77 POINT VALUE: 1.00

During a low reactor water level condition, Alternate Coolant Injection using demineralized water is being aligned using the HPCI system. A valid HPCI isolation signal is present, resulting in an automatic closure signal to the HPCI Injection valve (E41-F006).

How is the HPCI Injection Valve (E41-F006) opened to provide injection to the reactor?

E41-F006 is opened from the:

a. RTGB after placing the HPCI ASSD Interlock Defeat Switch on the RTGB to BYPASS.

b. MCC by placing the breaker's NORMAL/LOCAL switch to LOCAL to bypass valve interlocks.

c. RTGB and the breaker at the MCC is opened by an AO when the valve indicates full open.

d. RTGB after jumpers are installed to bypass the valve auto closure interlocks.

QUESTION 78 POINT VALUE: 1.00

Following a loss of drywell cooling, a small steam leak in the drywell results in the following containment conditions:

Drywell pressure	9 psig, rising
Suppression chamber pressure	8 psig, rising
Suppression pool level	+2 feet
Average Drywell temp	270°F, rising

The crew is directed to initiate drywell spray to control drywell temperature. Under current plant conditions, drywell spray may:

a. be initiated, all required conditions are met.

b. NOT be initiated, suppression pool level is too high.

c. NOT be initiated, suppression chamber pressure is too low.

d. NOT be initiated, conditions are in the UNSAFE region of the Drywell Spray Initiation Limit.

QUESTION 79 POINT VALUE: 1.00

The Suppression Chamber Spray Initiation Pressure is ________ in the Suppression Chamber and is based on:

a. 2.7 psig; intrusion of air into primary containment due to Reactor Building-Torus vacuum breaker operation.

b. 2.7 psig; the lowest suppression chamber pressure that RHR system logic will allow sprays to be initiated.

c. 13 psig; 95% of the noncondensibles in the drywell have been transferred to the suppression chamber airspace.

d. 13 psig; the highest pressure that initiation of sprays will prevent exceeding the Pressure Suppression Pressure Limit.

QUESTION 80 POINT VALUE: 1.00

During an accident on Unit One (1), the following primary containment and plant conditions exist:

Reactor pressure	798 psig
Suppression pool level	-42 inches
Suppression pool temperature	171°F
Suppression chamber pressure	17 psig

Current conditions are in the:

a. SAFE region of all Containment Limits.

b. UNSAFE region of the Heat Capacity Level Limit.

c. UNSAFE region of the Heat Capacity Temperature Limit.

d. UNSAFE region of the Pressure Suppression Pressure Limit.

QUESTION 81 POINT VALUE: 1.00

A primary system discharging into Secondary Containment has resulted in one area exceeding the Maximum Safe Operating Radiation Level, but within the EQ envelop. The radiation level in this area is subsequently reduced below the Maximum Safe value.

A second area subsequently exceeds its Maximum Safe Operating Radiation Level. What action is required by the Secondary Containment Control Procedure?

a. Shutdown the Reactor per GP-05.

b. Scram the Reactor and initiate a cooldown ≤100°F/Hour.

c. Scram the Reactor and initiate a cooldown >100°F/Hour.

d. Scram the Reactor and open seven ADS valves.

QUESTION 82 POINT VALUE: 1.00

While performing PT 10.1.1, RCIC OPERABILITY TEST, the RCIC steam supply line ruptured. RCIC failed to automatically isolate and attempts to manually isolate RCIC are unsuccessful.

The following Steam Leak Detection NUMAC channels are in alarm:

B21-XY-5949A, Channel A1-3, reading 298°F
B21-XY-5949B, Channel A1-3, reading 294°F
B21-XY-5948B, Channel A5-4, reading 301°F
B21-XY-5948A, Channel A5-4, reading 303°F

No other channels are in alarm. What action is <u>required</u> to be be taken?

a. Scram the reactor and emergency depressurize.

b. Scram the reactor and commence a cooldown at normal rates.

c. Scram the reactor and rapidly depressurize to the main condenser.

d. Shutdown the reactor using GP-05 or scram the reactor as directed by the Shift Supervisor.

QUESTION 83 POINT VALUE: 1.00

Unit Two (2) is operating at power when a rupture of RWCU piping downstream of the Non Regenerative Heat Exchangers occurs. RWCU Inboard Isolation Valve (G31-F001) and Outboard Isolation Valve (G31-F004) BOTH fail in the open position. Plant conditions:

Rx Bldg 50' temp	135°F
Rx Bldg 20' temp	105°F
S Core Spray Room	Flood Level Hi Hi alarm sealed in
S RHR Room	Flood Level Hi alarm sealed in

The operating crew is required to enter EOP-03-SCCP and:

a. continue attempts to isolate the leak, commence an immediate plant shutdown per GP-05.

b. continue attempts to isolate the leak, scram the reactor when 50' temperature exceeds 140°F.

c. immediately scram the reactor and consider anticipation of emergency depressurization.

d. immediately scram the reactor and open seven ADS valves for emergency depressurization.

QUESTION 84 POINT VALUE: 1.00

Following core damage, an unisolable steam leak in the Turbine Building requires declaration of a General Emergency due to loss of three out of three fission product barriers.

The crew is executing EOP-04-RRCP, Radiation Release Control Procedure. Field surveys and Off-Site dose projections (PEP-03.4.7) are being performed.

When, per EOP-04-RRCP, is Emergency Depressurization of the Reactor required to be initiated?

a. Immediately since a General Emergency has been declared.

b. If the Off-Site release rate exceeds the Emergency Action Level for an Alert.

c. If the Off-Site release rate approaches or exceeds the Emergency Action Level for a Site Area Emergency.

d. If the Off-Site release rate approaches or exceeds the Emergency Action Level for a General Emergency.

QUESTION 85 POINT VALUE: 1.00

During emergency conditions, it has been determined that exposures in excess of 10CFR20 limits may be required to protect equipment important to reactor safety which is needed to protect the population of Brunswick County from a large release.

In accordance with PEP-03.7.6, the Emergency Worker Dose Limit is:

a. 10 Rem TEDE. This dose may be authorized by the Site Emergency Coordinator.

b. 10 Rem TEDE. This dose shall be authorized by the Radiological Controls Manager.

c. 25 Rem TEDE. This dose may be authorized by the Site Emergency Coordinator.

d. 25 Rem TEDE. This dose shall be authorized by the Radiological Controls Manager.

QUESTION 86 POINT VALUE: 1.00

A Unit Two (2) Reactor startup is in progress per GP-02. Heatup and pressurization of the Reactor is being performed.

The operating CRD Pump trips. Attempts to restart CRD per OP-08 and AOP-02.0 are unsuccessful.

AOP-02.0 requires the operator to insert a manual Reactor scram only if Reactor pressure is below:

a. 200 psig

b. 400 psig

c. 600 psig

d. 800 psig

QUESTION 87 POINT VALUE: 1.00

Unit One (1) is operating at 100% power when Recirculation Pump 1B trips, resulting in the following conditions:

Total Core Flow (P603)	39 Mlbm/Hour
Total Core Flow (U1CPWTCF)	35 Mlbm/Hour
Indicated Core Plate DP	4.7 psid
APRMs	68%
LPRM Upscale/Downscale alarms	None

What region of the Thermal Power Limitations Map is the plant operating in, and what operator action is required to be taken?

a. Region B, raise total core flow.

b. Region B, insert control rods per ENP-24.

c. Region A, immediately insert a manual scram.

d. 5% Buffer, increase monitoring of nuclear instrumentation.

QUESTION 88 POINT VALUE: 1.00

Unit Two (2) was operating at power when a trip and lockout of BOP bus 2B required the operator to insert a manual Reactor scram. Shortly following the scram, the following indications are noted:

Recirc pump A #1 seal pressure	1000 psig
Recirc pump A #2 seal pressure	1000 psig
Recirc pump B #1 seal pressure	100 psig
Recirc pump B #2 seal pressure	50 psig
Drywell pressure	1.4 psig, rising
Average drywell temp	140°F, rising
Average primary containment temp	126°F, rising

The operator is required to enter:

a. AOP-14.0 and isolate Recirc pump A.

b. AOP-14.0 and isolate Recirc pump B.

c. EOP-02-PCCP and isolate Recirc pump A.

d. EOP-02-PCCP and isolate Recirc pump B.

QUESTION 89 POINT VALUE: 1.00

A situation arises requiring immediate evacuation of the control room prior to completion of any immediate actions per AOP-32.0. RPS is aligned:

RPS Bus A	Powered from RPS MG Set A
RPS Bus B	Powered from RPS MG Set B

If the RPS EPA breakers are opened in the exact sequence specified by AOP-32.0, opening which EPA breaker will result in a reactor scram?

a. EPA Breaker 1.

b. EPA Breaker 2.

c. EPA Breaker 3.

d. EPA Breaker 4.

QUESTION 90 POINT VALUE: 1.00

Following a loss of feedwater on Unit Two (2), HPCI and RCIC are being used to restore Reactor water level to the normal band. The operator notes the following alarms and indications:

250 Batt B Under Voltage	Alarm sealed in
Battery Bus 2B-1 Voltage	0 volts (XU-2)
Battery Bus 2B-2 Voltage	0 volts (XU-2)
Battery Bus 2B-1 Voltage	0 volts (ERFIS)
Battery Bus 2B-2 Voltage	0 volts (ERFIS)

How is the operation of HPCI and RCIC affected by the power loss?

a. HPCI continues to inject to the Reactor, RCIC isolates due to loss of isolation logic power.

b. RCIC continues to inject to the Reactor, HPCI isolates due to loss of isolation logic power.

c. HPCI continues to inject to the Reactor, RCIC coasts down due to loss of flow controller power.

d. RCIC continues to inject to the Reactor, HPCI coasts down due to loss of flow controller power.

QUESTION 91 POINT VALUE: 1.00

Following a Loss of Off-Site Power to Unit One (1), the operator is performing AOP-36.1. Plant conditions are:

Diesel Generator 1	Running at 3575 KW load
Diesel Generator 2	Running at 3680 KW load
Reactor Building HVAC	Isolated

The operator is directed to restart Reactor Building HVAC using three (3) supply fans (75 KW each) and three (3) exhaust fans (45 KW each).

How will starting two supply and exhaust fans from MCC 1XG and one supply and exhaust fan from MCC 1XH affect Diesel Generator (DG) maximum loading?

a. DG1 only maximum load will be exceeded.

b. DG2 only maximum load will be exceeded.

c. DG1 and DG2 maximum load will be exceeded.

d. DG1 and DG2 will remain within maximum load limits.

QUESTION 92 POINT VALUE: 1.00

Following a Loss Of Off-Site Power on Unit One (1), Diesel Generators #1 and #2 are tied to their respective Emergency Buses. A rupture of the Unit One (1) Nuclear Service Water Header in the Service Water Building results in the following indications:

Nuclear Header Pressure (XU-2)	0 psig
Nuclear Hdr Serv Wtr Press-Low	Alarm sealed in

Diesel Generators #1 and #2:

a. have no available cooling and will trip.

b. have no available cooling but will continue to run.

c. cooling water supply will automatically transfer to the Unit Two (2) Nuclear Service Water header.

d. cooling water supply will automatically transfer to the Unit One (1) Conventional Service Water header.

QUESTION 93 POINT VALUE: 1.00

Unit Two (2) was at 20% power during plant startup when a sudden rise in off gas flow is accompanied by a lowering condenser vacuum. The reactor was manually scrammed. Plant conditions:

Condenser vacuum	15" Hg, slowly lowering
Reactor pressure	921 psig, steady
Bypass valves	One partially open

What is the minimum additional reduction in condenser vacuum that would result in the loss of automatic Reactor pressure control?

a. 5" Hg.

b. 6" Hg.

c. 7" Hg.

d. 8" Hg.

QUESTION 94 POINT VALUE: 1.00

Unit One (1) is performing refueling operations. A fuel handling accident results the the following radiation alarms:

Area Rad Refuel Floor Hi (white alarm)
Area Rad Rx Bldg Hi (red alarm, blue bar)
Area Rad Control Room Hi (red alarm, red bar)
Rx Bldg Vent Rad Hi (red alarm, blue bar)
Rx Bldg Vent Rad Hi Hi (red alarm, blue bar)

How will the Reactor Building HVAC and Control Building Emergency Air Filtration (CBEAF) systems respond to the above radiation alarms?

a. Reactor Building HVAC isolates, CBEAF remains in standby.

b. Reactor Building HVAC remains in operation, CBEAF initiates.

c. Reactor Building HVAC remains in operation, CBEAF remains in standby.

d. Reactor Building HVAC isolates and CBEAF initiates.

QUESTION 95 POINT VALUE: 1.00

During normal full power operation of Unit Two (2), the following alarms and indications are noted:

Air Compressor D Trip	Alarm sealed in
Air Compressors A/B/C	Running
Instrument Air Pressure low	Alarm sealed in
Instrument Air header pressure	100 psig

The operator should verify that air compressors A, B and C are loaded and that:

a. Service air isolation valves, PV-706-1 and PV-706-2, have automatically closed.

b. Interruptible air isolation valves, PV-722-1 and PV-722-2, have automatically closed.

c. Standby reactor building air compressors have automatically started and loaded.

d. Backup nitrogen rack isolation valves, RNA-SV-5482 and SV-5481, have automatically opened.

QUESTION 96 POINT VALUE: 1.00

Following a loss of shutdown cooling, Alternate Shutdown Cooling has been established per AOP-15.0. Plant conditions are:

RHR Loop A in suppression pool cooling
RHR Loop B injecting to the Reactor Vessel
Reactor pressure is 115 psig
SRV G is open

It becomes desired to make a slight adjustment to raise the cooldown rate. This may be accomplished by closing SRV G and opening:

a. SRV H.

b. SRV J.

c. SRV K.

d. SRV L.

QUESTION 97 POINT VALUE: 1.00

Unit Two (2) is operating with the following plant conditions:

Reactor power	85%
Core flow	51 Mlbm/hr
Rod line	110%
Recirc MG sets	Scoop tubes locked

Which of the following conditions authorizes the operator to manually initiate Select Rod Insert (SRI)?

a. Condenser vacuum lowers and approaches the turbine trip setpoint.

b. Feedwater heating is partially lost and APRMs approach the scram setpoint.

c. A reactor feed pump trips and reactor level approaches the scram setpoint.

d. A recirculation pump trips placing the plant in Region A of the Thermal Power Limitations Map.

QUESTION 98 POINT VALUE: 1.00

Following a loss of Off-Site Power to BOTH Units, the following conditions exist on Unit One (1):

DG 1	Tripped
DG 2	Tripped
HPCI	Unavailable
RCIC	Operating at rated flow

The operator is required by AOP-36.~~1~~ 2 to commence a Reactor cooldown at a rate:

a. greater than 100°F/hour to between 50 and 150 psig Reactor pressure.

b. greater than 100°F/hour to between 150 and 300 psig Reactor pressure.

c. as close as possible, but not to exceed 100°F/hour to between 50 and 150 psig Reactor pressure.

d. as close as possible, but not to exceed 100°F/hour to between 150 and 300 psig Reactor pressure.

QUESTION 99 POINT VALUE: 1.00

Core defueling is in progress. All control rods are fully inserted into the reactor core. A fuel assembly has just been placed in the fuel pool and unlatched. The main hoist has been raised to a safe elevation to pass through the cattle chute (**NOT** normal-up) with the bridge still over the fuel pool location.

The next step requires that a fuel assembly be removed from the reactor core and placed in the fuel pool.

When will the ROD BLOCK INTERLOCK #1 light on the Interlock Status Display Panel first light as the next step is performed?

a. As the bridge is moved near the reactor core (LS1 is actuated).

b. When the bridge is over the reactor core (LS1 is actuated) and the main hoist is lowered into the reactor vessel.

c. When the fuel assembly is latched, with both grapple hooks closed.

d. When the fuel assembly is being raised and the main hoist loaded signal is actuated.

QUESTION 100 POINT VALUE: 1.00

Core Spray Pump 2A 4160 volt breaker is racked in per OP-50, with 125V DC available at the switchgear. A LOCA results in a condition requiring the auto start of the pump.

Refer to LL-09113 sheet 15, Core Spray Pump 2A Control Wiring Diagram. The pump breaker is closed by energizing the:

a. X coil, when relay K12A is de-energized and relay K15A is energized.

b. Y coil, when relay K12A is de-energized and relay K15A is energized.

c. X coil, when relay K12A is energized and relay K15A is de-energized.

d. Y coil, when relay K12A is energized and relay K15A is de-energized.

** END OF "NRC 97-1 RO, Rev 0" EXAMINATION **

NAME: RO Answer Key DATE: / / SCORE:

GRADED BY: ALTERNATE GRADER: (if required)

EXAM: NRC 97-1 RO, Rev 0 CLASS: HLC 96-1 COURSE CODE: ROA01B

1. A B C D	21. A B C D	41. A B C D	61. A B C D
2. A B C D	22. A B C D	42. A B C D	62. A B C D
3. A B C D	23. A B C D	43. A B C D	63. A B C D
4. A B C D	24. A B C D	44. A B C D	64. A B C D
5. A B C D	25. A B C D	45. A B C D	65. A B C D
6. A B C D	26. A B C D	46. A B C D	66. A B C D
7. A B C D	27. A B C D	47. A B C D	67. A B C D
8. A B C D	28. A B C D	48. A B C D	68. A B C D
9. A B C D	29. A B C D	49. A B C D	69. A B C D
10. A B C D	30. A B C D	50. A B C D	70. A B C D
11. A B C D	31. A B C D	51. A B C D	71. A B C D
12. A B C D	32. A B C D	52. A B C D	72. A B C D
13. A B C D	33. A B C D	53. A B C D	73. A B C D
14. A B C D	34. A B C D	54. A B C D	74. A B C D
15. A B C D	35. A B C D	55. A B C D	75. A B C D
16. A B C D	36. A B C D	56. A B C D	76. A B C D
17. A B C D	37. A B C D	57. A B C D	77. A B C D
18. A B C D	38. A B C D	58. A B C D	78. A B C D
19. A B C D	39. A B C D	59. A B C D	79. A B C D
20. A B C D	40. A B C D	60. A B C D	80. A B C D

NAME: RO Answer Key DATE: / / SCORE:

GRADED BY: ALTERNATE GRADER: (if required)

EXAM: NRC 97-1 RO, Rev 0 CLASS: HLC 96-1 COURSE CODE: ROA01B

81. A B C D
82. A B C D
83. A B C D
84. A B C D
85. A B C D
86. A B C D
87. A B C D
88. A B C D
89. A B C D
90. A B C D
91. A B C D
92. A B C D
93. A B C D
94. A B C D
95. A B C D
96. A B C D
97. A B C D
98. A B C D
99. A B C D
100. A B C D

REFERENCES

FOR WRITTEN TEST

RO & SRO

EOP-01-UG
Attachment 6
Reactor Water Level Caution
(Caution 1)

ATTACHMENT 6
REACTOR WATER LEVEL CAUTION
(Caution 1)

A reactor water level instrument may be used to determine reactor water level only when the conditions for use as listed in Table 1 are satisfied for that instrument.

TABLE 1

CONDITIONS FOR USE OF REACTOR WATER LEVEL INSTRUMENTS

NOTE

Reference leg area drywell temperature is determined using Figure 13, ERFIS, or Instructional Aid based on Figure 13.

NOTE

Immediate reference leg boiling is not expected to occur for short duration excursions into the unsafe region due to heating of the drywell. The thermal time constant associated with the mass of metal and water in the reference leg will prohibit immediate boiling of the reference leg. Reference leg boiling is an obvious phenomenon. Large scale oscillations of all water level instruments associated with the reference leg that is boiling will occur. This occurrence will be obvious and readily observable by the operator. Additionally, if the operator is not certain whether boiling has occurred, he can refer to plant history as provided on water level recorders or ERFIS. Reference leg boiling is indicated by level oscillations without corresponding pressure oscillations. If no boiling of the reference legs occurs and drywell temperature and pressure are restored to the safe region, then the instrument in question should continue to be used.

Instrument	Conditions for Use
Narrow Range Level Instruments C32-LI-R606A, B, C (N004A, B, C) C32-LPR-R608 (N004A, B) Indicating Range 150-210 Inches Cold Reference Leg	The reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14) AND Unit 1 only: The indicated level is in the SAFE region of Figure 15. Unit 2 only: The indicated level is in the SAFE region of Figure 15A.
Shutdown Range Level Instruments B21-LI-R605A, B (N027A, B) Indicating Range 150-550 Inches Cold Reference Leg	The reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14) AND Unit 1 only: The indicated level is in the SAFE region of Figure 16. Unit 2 only: The indicated level is in the SAFE region of Figure 16A.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Wide Range Level Instruments B21-LI-R604A, B (N026A, B) C32-PR-R609 (N026B) Indicating Range 0-210 Inches Cold Reference Leg	1. Temperature on the Reactor Building 50' below 140°F (B21-XY-5948A A2-4, B21-XY-5948B A2-4, ERFIS Computer Point B21TA102, <u>OR</u> B21TA103) <u>AND</u> 2. <u>IF</u> the reference leg area drywell temperature is in the UNSAFE region of the Reactor Saturation Limit (Figure 14), <u>THEN</u> the indicated level is greater than 20 inches <u>OR</u> <u>IF</u> the reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), <u>THEN</u> the indicated level is greater than 10 inches.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
Fuel Zone Level Instruments B21-LI-R610 (N036) B21-LR-R615 (N037) Indicating Range -150 - +150 Inches Cold Reference Leg	1. The reference leg area drywell temperature is in the SAFE region of the Reactor Saturation Limit (Figure 14), AND 2. IF the reference leg area drywell temperature is less than 440°F, THEN the indicated level is greater than -150 inches OR IF the reference leg area drywell temperature is greater than or equal to 440°F, THEN the indicated level is greater than -130 inches. AND 3. Reactor Recirculation Pumps are shutdown. NOTE To determine reactor water level at TAF, see Unit 1 only: Figure 17 and Unit 2 only: Figure 17A To determine reactor water level at the minimum steam cooling level (LL-4), see Unit 1 only: Figure 18 and Unit 2 only: Figure 18A To determine reactor water level at the minimum zero injection level (LL-5), see Unit 1 only: Figure 19 and Unit 2 only: Figure 19A Continued on next page.

ATTACHMENT 6 (Cont'd)

TABLE 1 (Cont'd)

Instrument	Conditions for Use
	NOTE Each figure has two curves: The upper curve for reference leg area drywell temperature greater than 200°F. The lower curve for reference leg area drywell temperature less than or equal to 200°F. If containment conditions are such that reference leg area temperatures could not be controlled and maintained less than the 200°F requirement, then the upper lines on the graph should be utilized. NOTE These level instruments are valid for indication with RHR LPCI flow.

ATTACHMENT 6 (Cont'd)

FIGURE 13
LEVEL INSTRUMENT REFERENCE LEG AREA
DRYWELL TEMPERATURE CALCULATIONS

1. For all Level Instruments EXCEPT B21-LI-R605 A, B, (N027 A, B); the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1B Point A ______
CAC-TR-4426-1B Point B ______
CAC-TR-4426-2B Point A ______
CAC-TR-4426-2B Point B ______

OR

Microprocessor

CAC-TY-4426-1 Point 5801 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

2. For Level Instruments B21-LI-R605A, B (N027A, B), the reference leg area drywell temperature is the highest of the following points:

Recorder

CAC-TR-4426-1A Point D ______
CAC-TR-4426-1B Point B ______
CAC-TR-4426-2A Point C ______
CAC-TR-4426-2A Point D ______
CAC-TR-4426-2B Point A ______
CAC-TR-4426-2B Point B ______

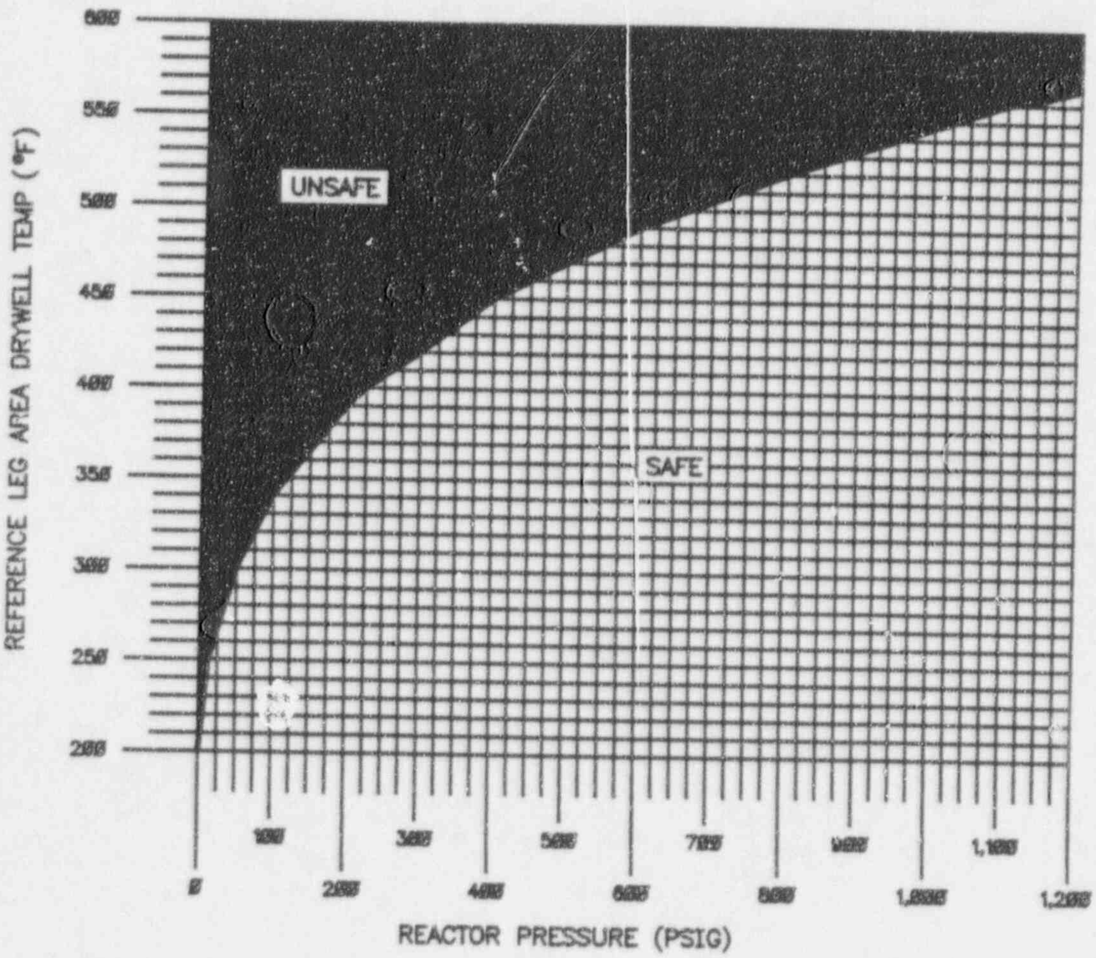
OR

Microprocessor

CAC-TY-4426-1 Point 5822 ______
CAC-TY-4426-1 Point 5803 ______
CAC-TY-4426-2 Point 5823 ______
CAC-TY-4426-2 Point 5824 ______
CAC-TY-4426-2 Point 5802 ______
CAC-TY-4426-2 Point 5804 ______

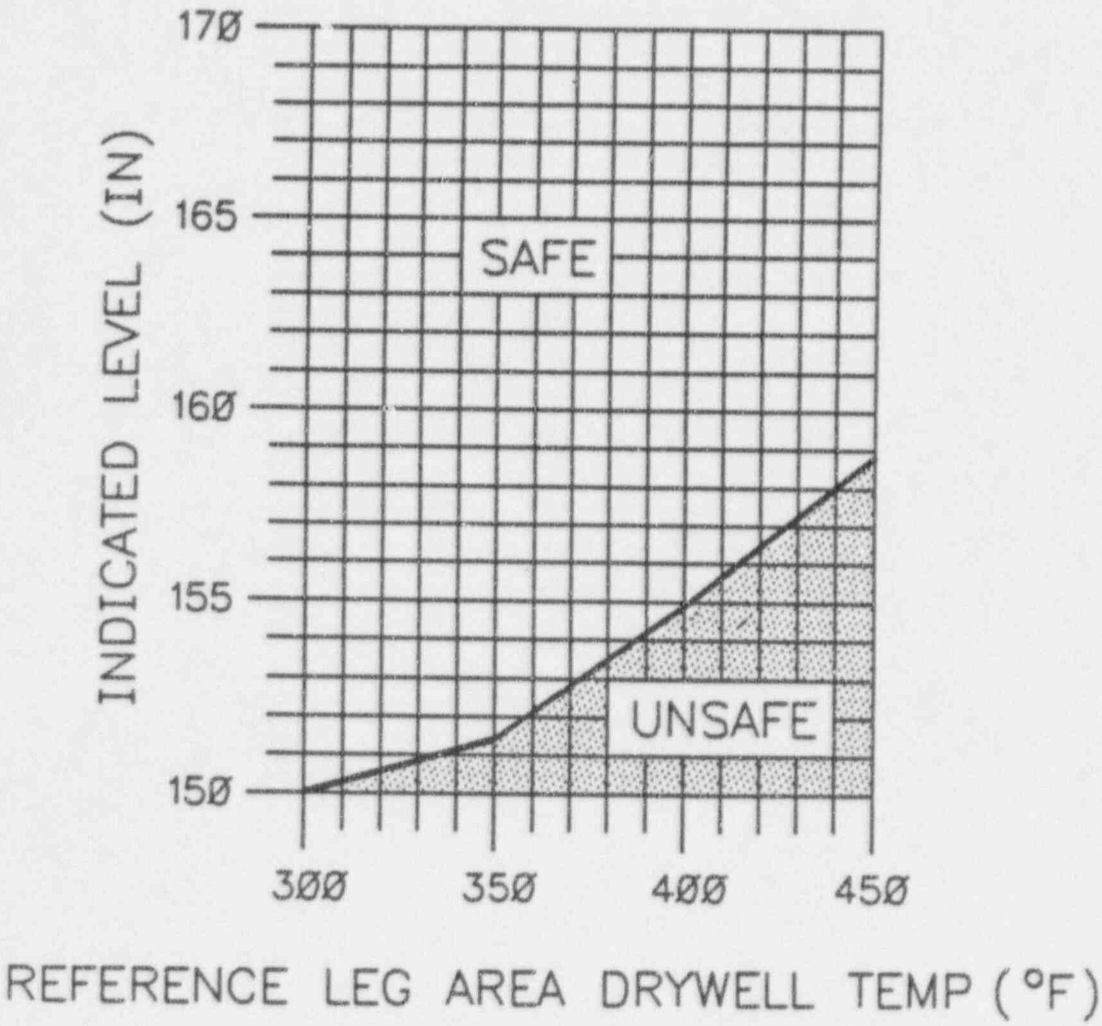
ATTACHMENT 6 (Cont'd)

FIGURE 14
REACTOR SATURATION LIMIT



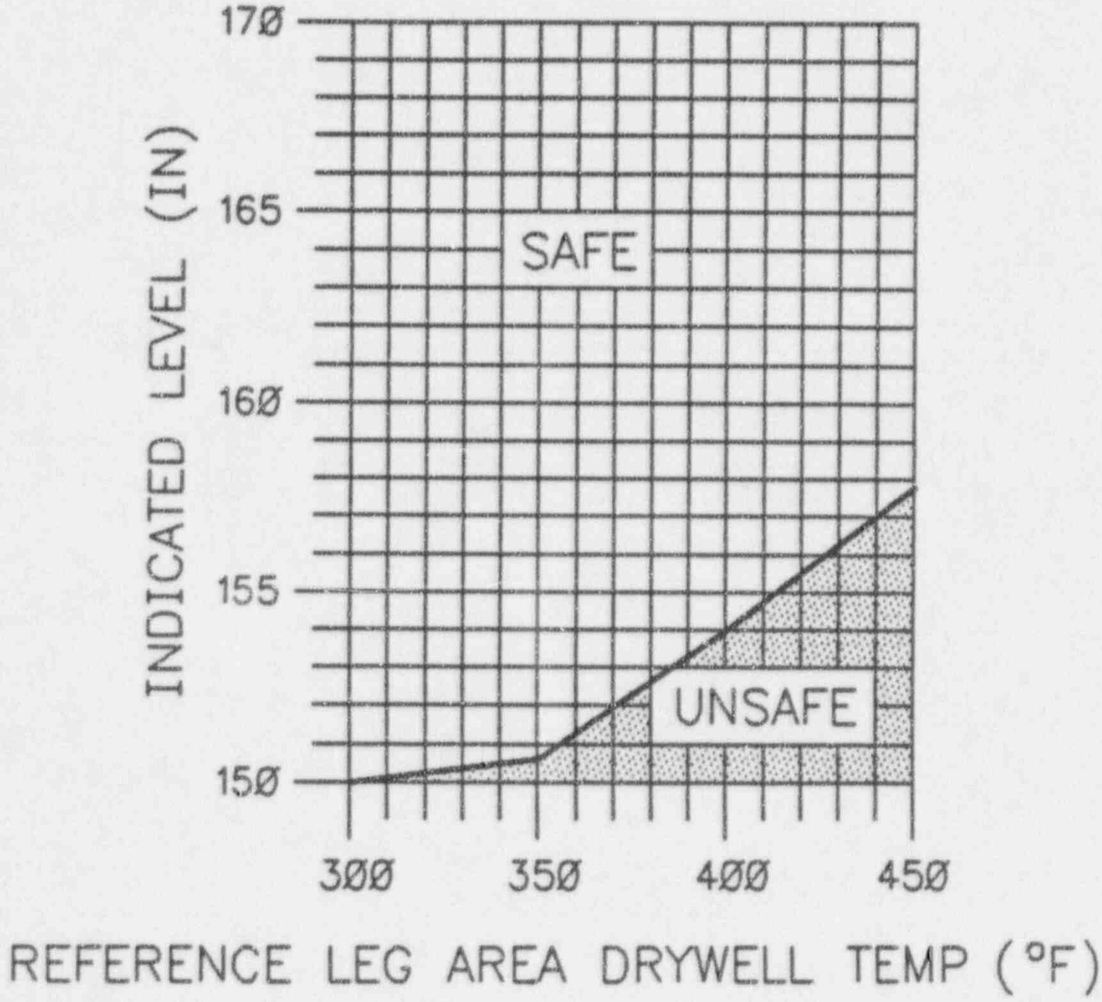
ATTACHMENT 6 (Cont'd)

FIGURE 15
UNIT 1 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



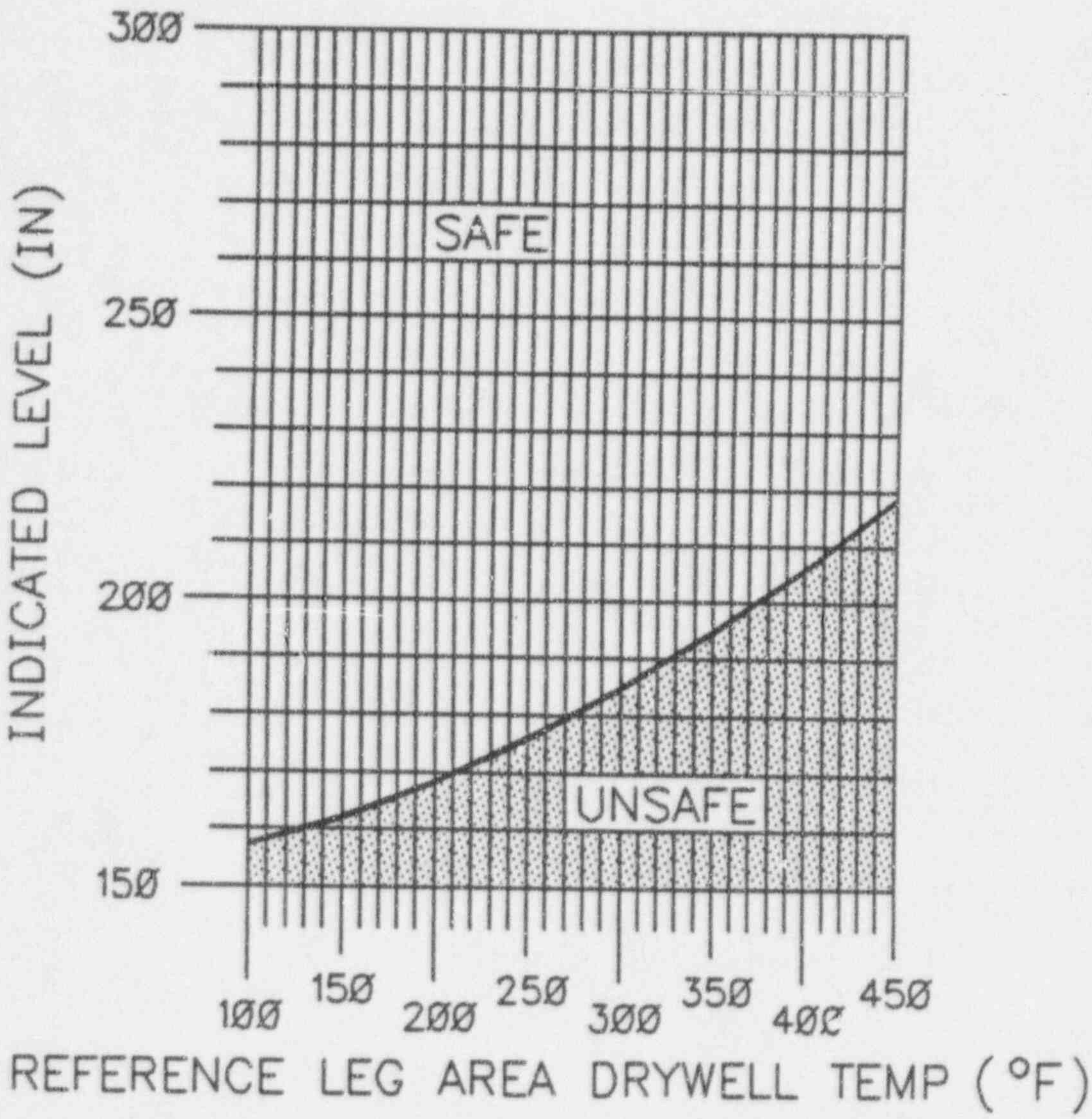
ATTACHMENT 6 (Cont'd)

FIGURE 15A
UNIT 2 NARROW RANGE LEVEL
INSTRUMENT (N004A, B, C) CAUTION



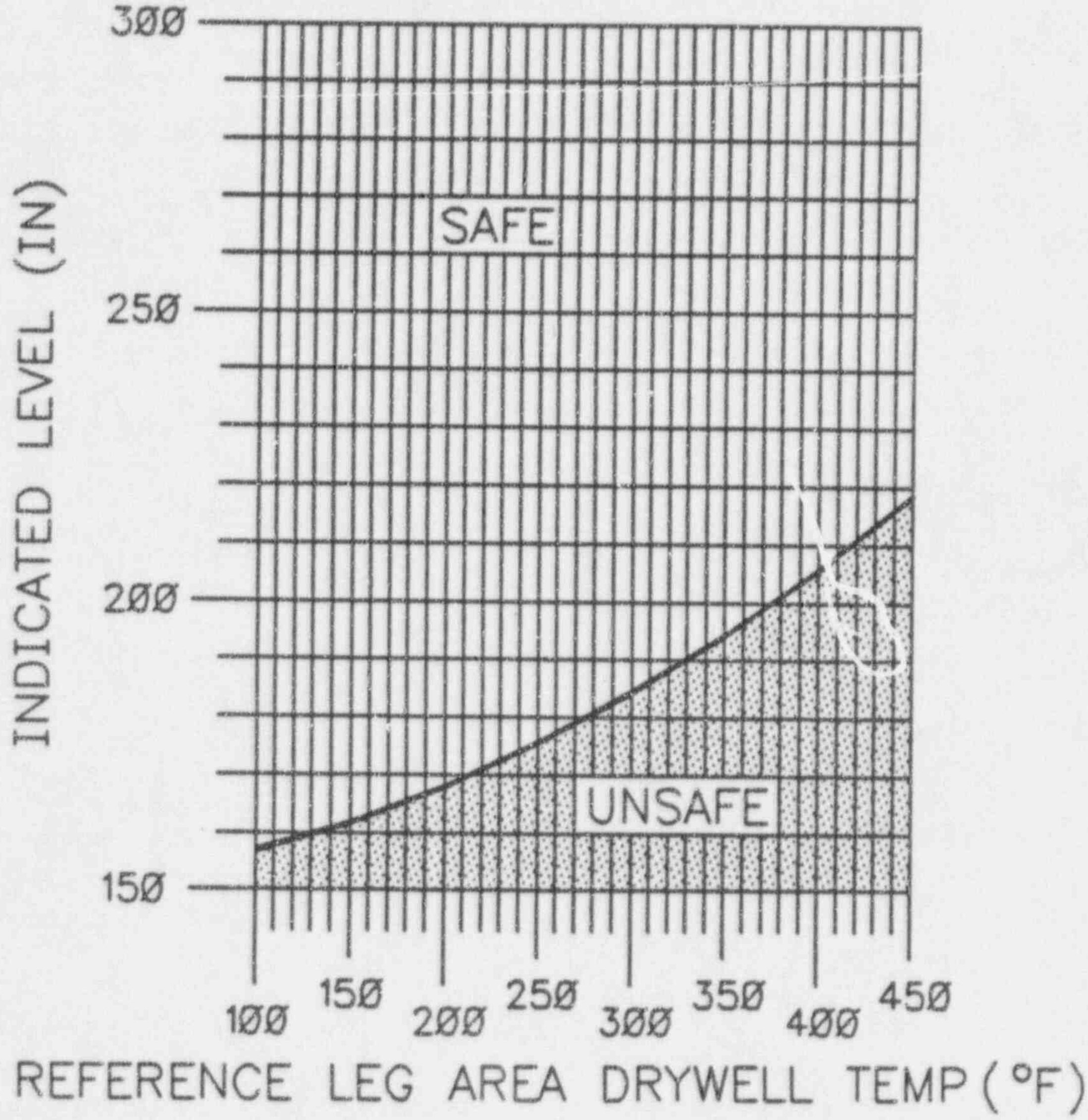
ATTACHMENT 6 (Cont'd)

FIGURE 16
UNIT 1 SHUTDOWN RANGE LEVEL
INSTRUMENT (N027A, B) CAUTION



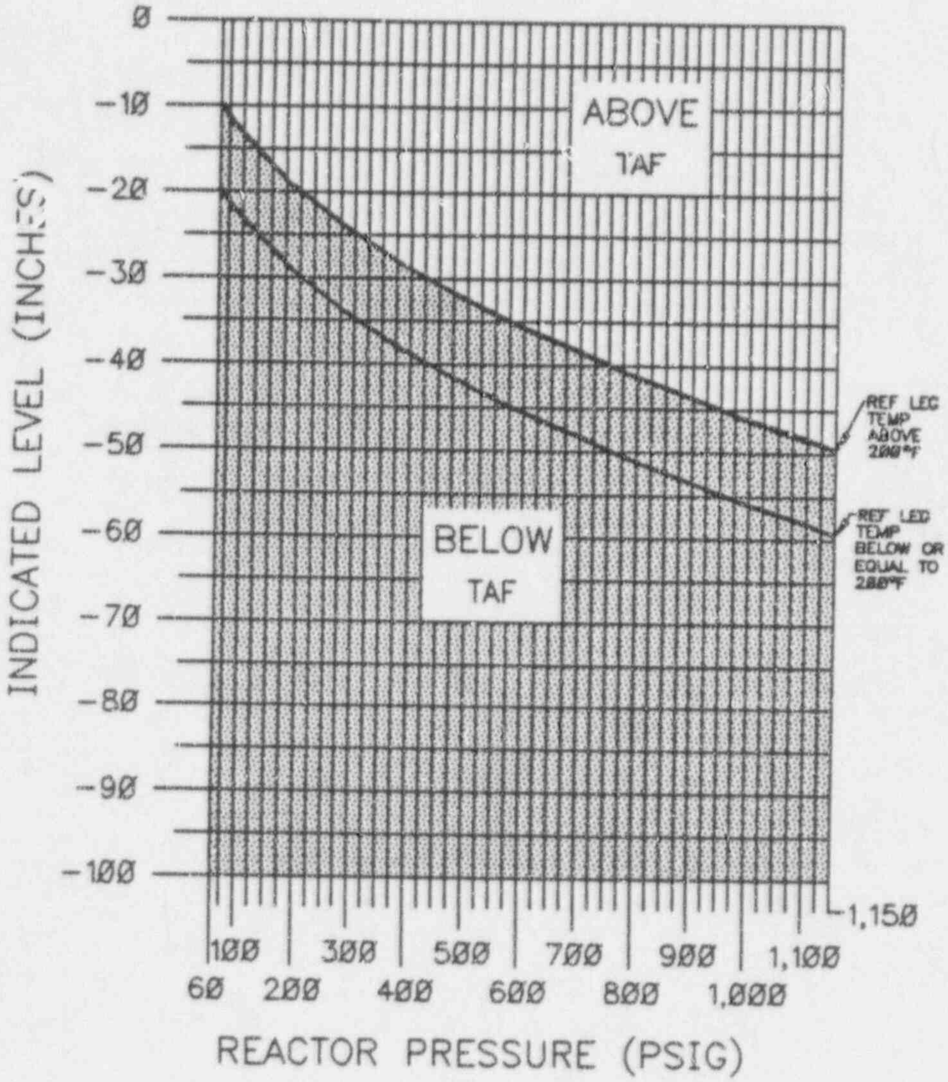
ATTACHMENT 6 (Cont'd)

FIGURE 16A
UNIT 2 SHUTDOWN RANGE LEVEL
INSTRUMENT (N027A, B) CAUTION



ATTACHMENT 6 (Cont'd)

FIGURE 17
UNIT 1 REACTOR WATER LEVEL AT TAF

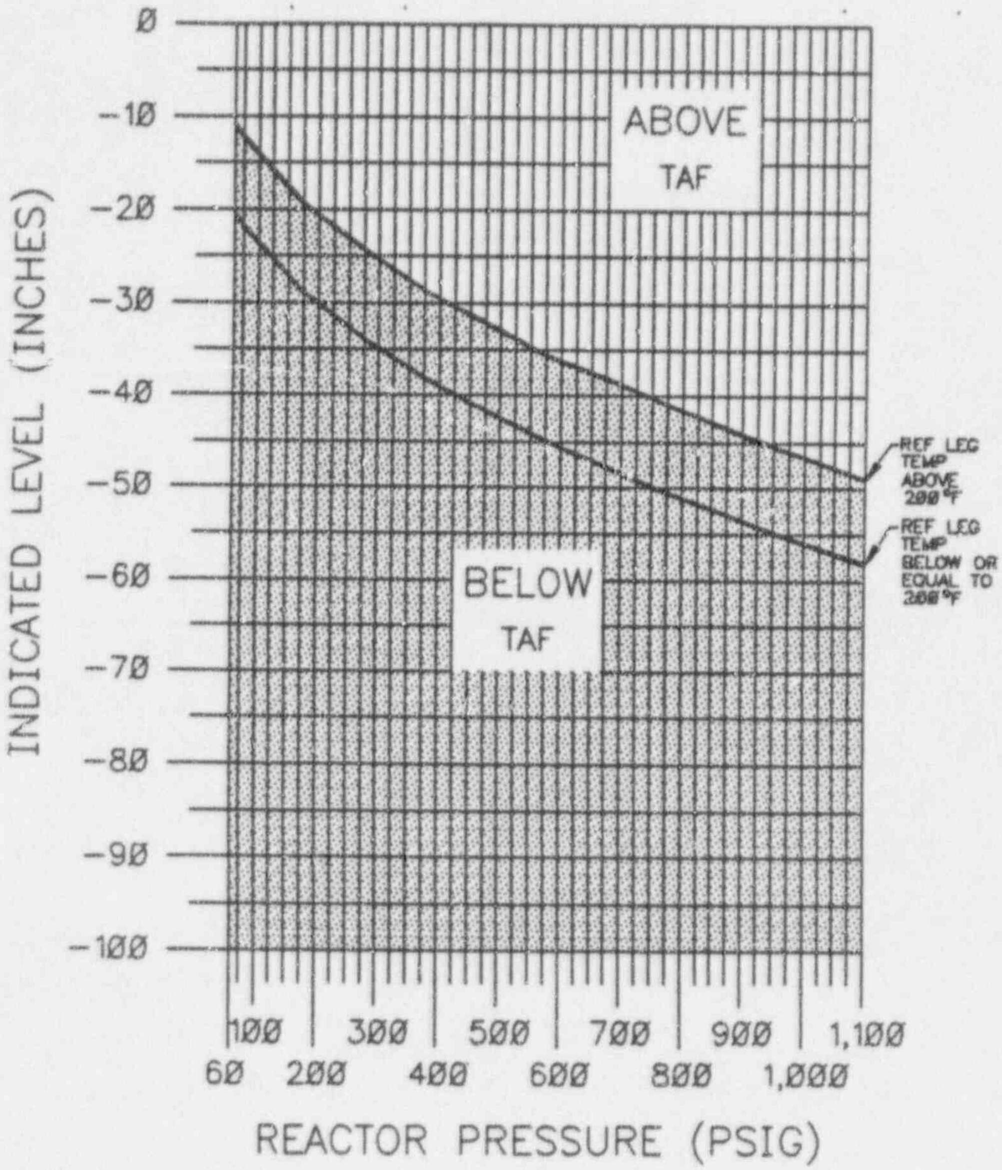


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 17A
UNIT 2 REACTOR WATER LEVEL AT TAF

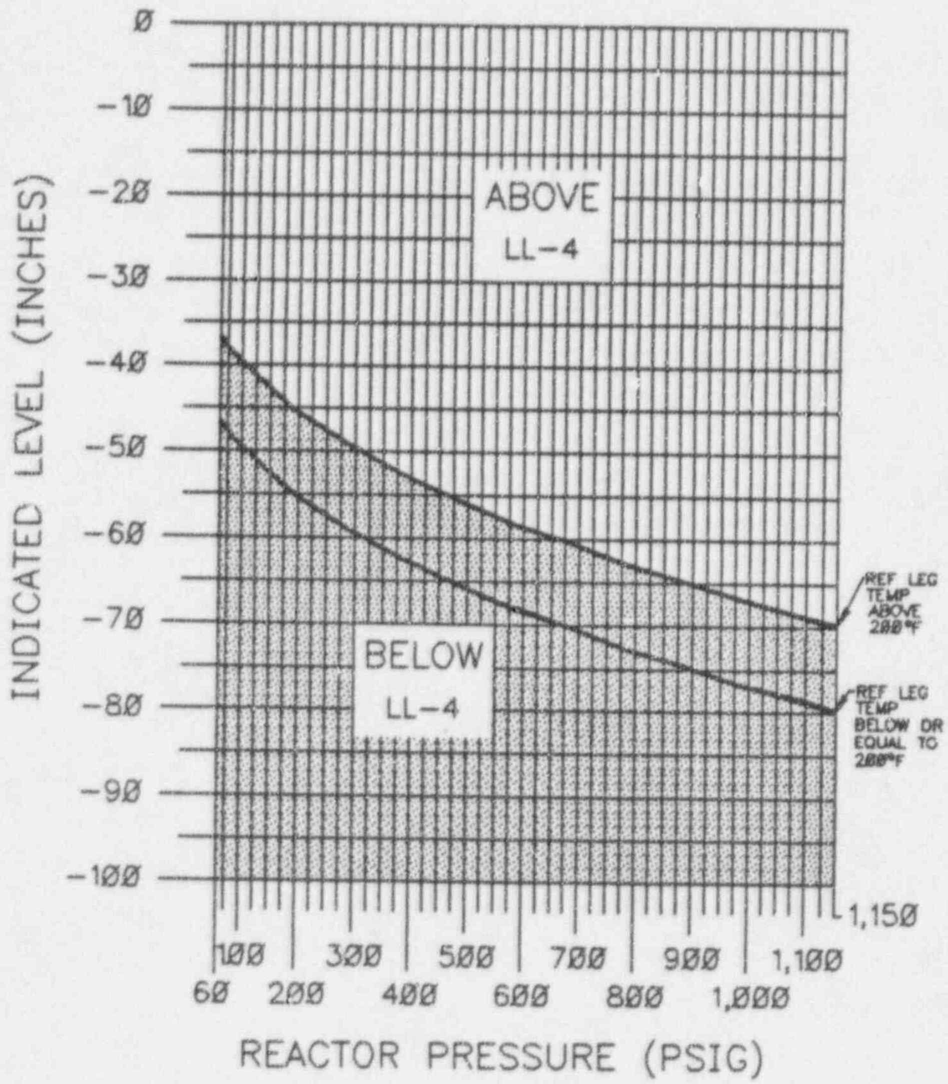


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
TAF IS -7.5 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18
UNIT 1 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

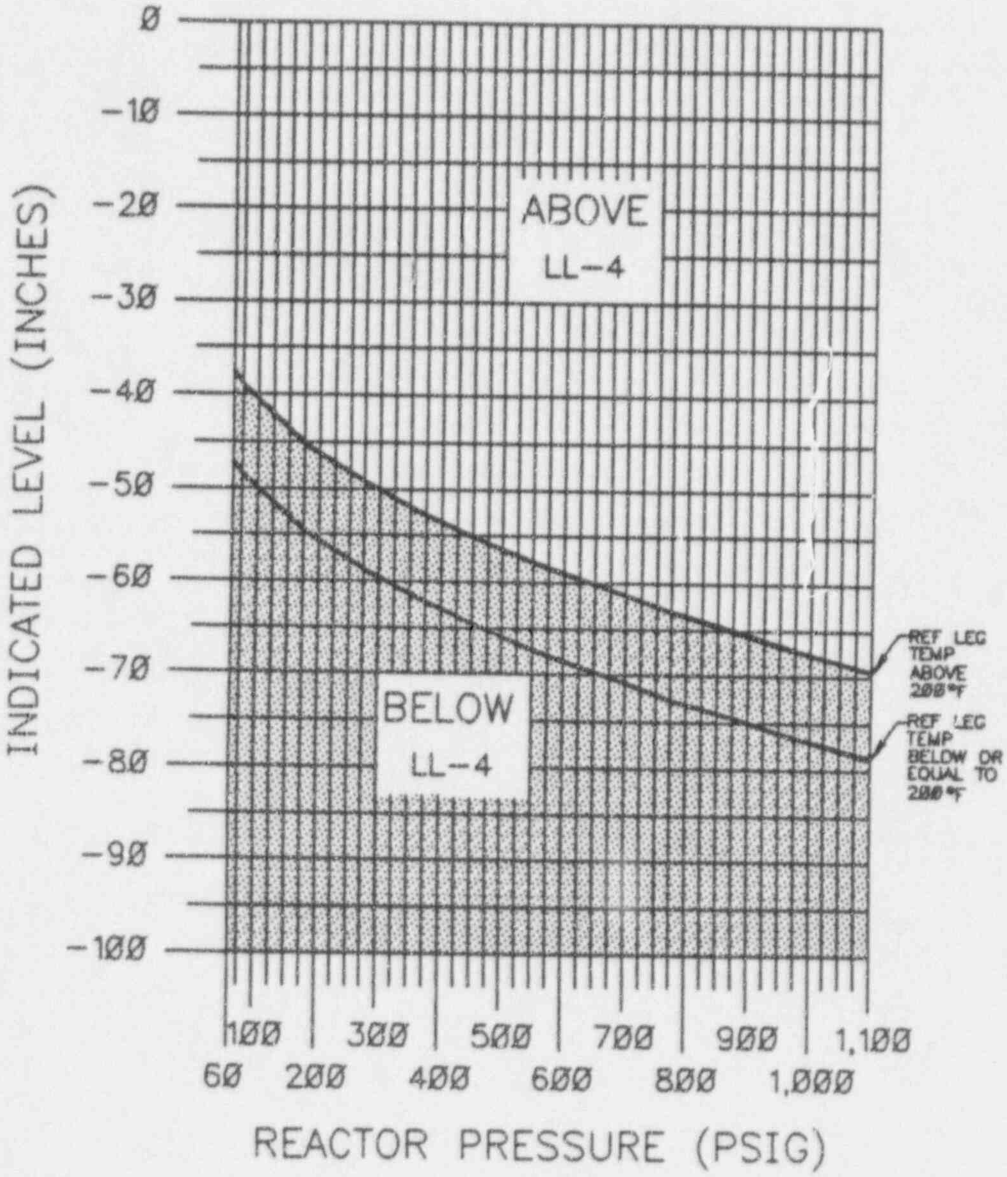


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -35 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 18A
UNIT 2 REACTOR WATER LEVEL AT LL-4
(MINIMUM STEAM COOLING LEVEL)

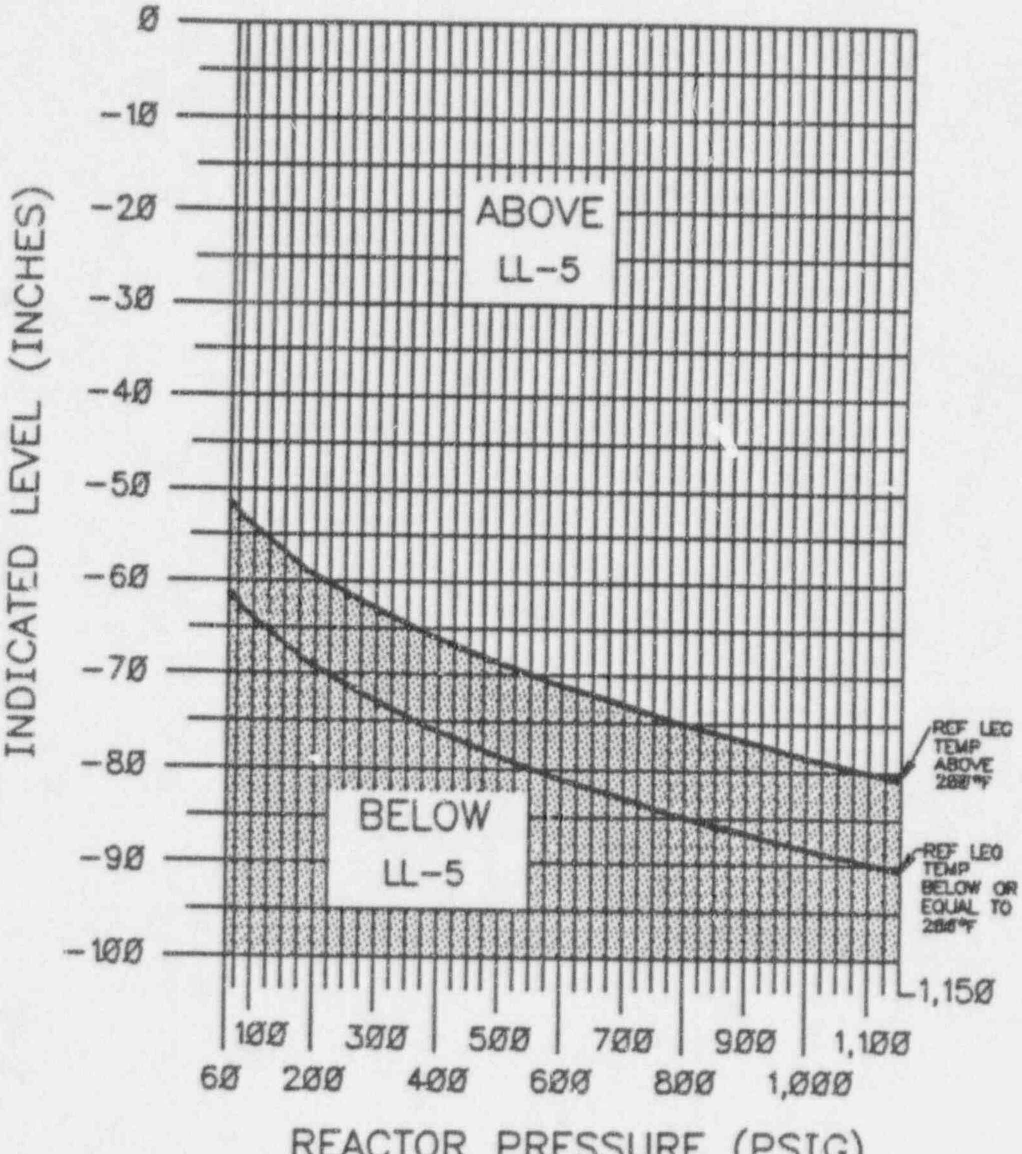


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-4 IS -35 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19
UNIT 1 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)

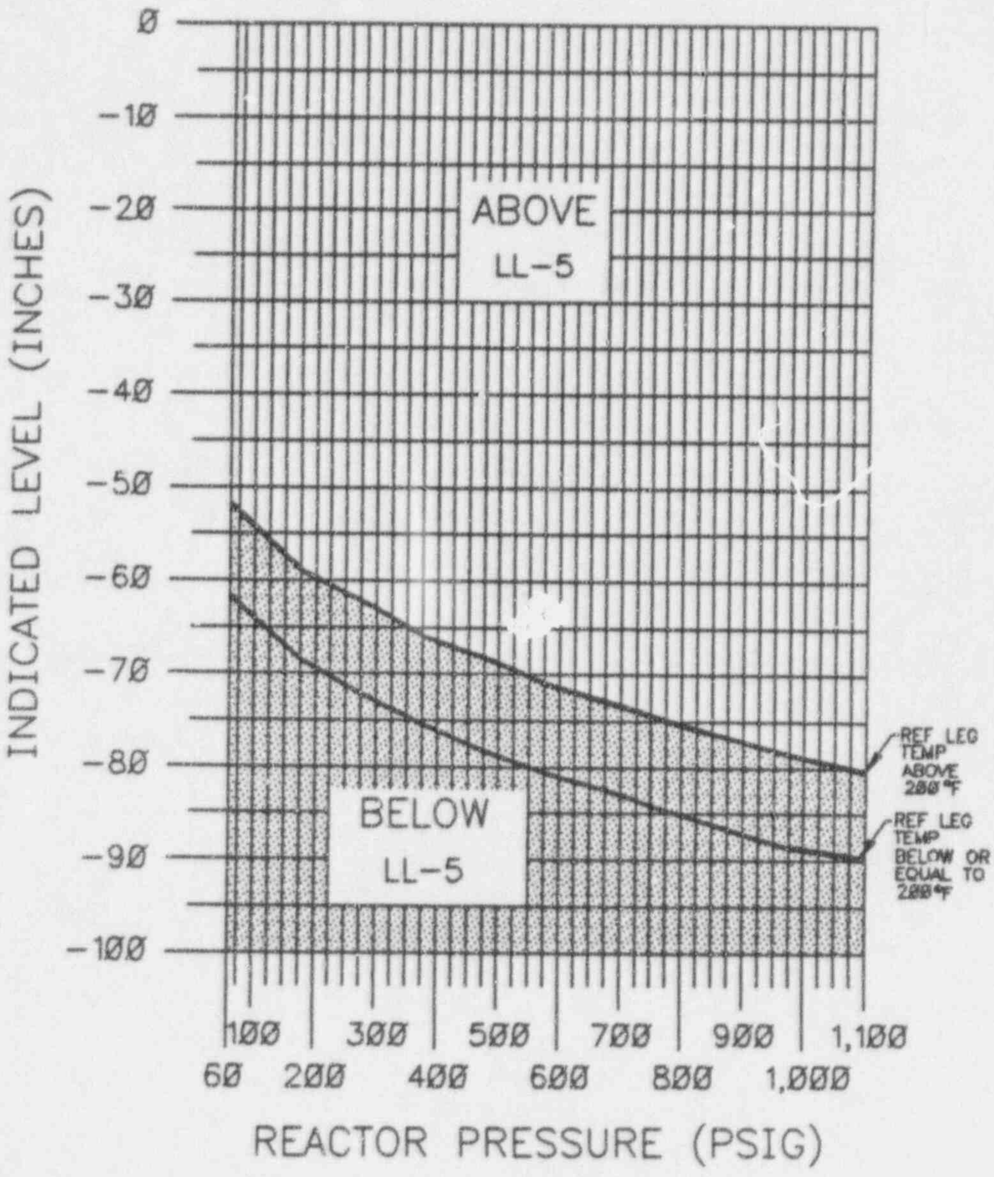


NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-5 IS -50 INCHES.

ATTACHMENT 6 (Cont'd)

FIGURE 19A
UNIT 2 REACTOR WATER LEVEL AT LL-5
(MINIMUM ZERO INJECTION LEVEL)



NOTE

WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
LL-5 IS -50 INCHES.

EOP-01-UG
Attachment 10
Secondary Containment Temperature
And Radiation Limits

ATTACHMENT 10
SECONDARY CONTAINMENT TEMPERATURE AND RADIATION LIMITS

FIGURE 20
SECONDARY CONTAINMENT AREA TEMPERATURE

TABLE 1
AREA TEMPERATURE LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	STEAM LEAK DETECTION CHANNEL/LOCATION	INSTRUMENT NUMBER/ WINDOW	MAX NORM OPERATING VALUE (°F) (NOTE 1)	MAX SAFE OPERATING VALUE (°F)	AUTO GROUP ISOL
N CORE SPRAY	N CORE SPRAY ROOM	PANEL XU-3	VA-TI-1603	120	175	N/A
S CORE SPRAY	S CORE SPRAY ROOM	PANEL XU-3	VA-TI-1604	120	175	N/A
RWCU	RWCU PUMP ROOM A	B21-XY-5949A B21-XY-5949B CH. A1-1	G31-TE-N016A G31-TE-N016B	140	225	3
	RWCU PUMP ROOM B	B21-XY-5949A B21-XY-5949B CH. A2-1	G31-TE-N016C G31-TE-N016D			
	RWCU HX ROOM	B21-XY-5949A B21-XY-5949B CH. A3-1	G31-TE-N016E G31-TE-N016F			
N RHR	N RHR EQUIP ROOM	B21-XY-5948A CH. A5-4 PANEL XU-3	E11-TE-N009A VA-TI-1601	175	295	N/A
S RHR	S RHR EQUIP ROOM	B21-XY-5948B CH. A5-4 PANEL XU-3	E11-TE-N009B VA-TI-1602	175	295	N/A
	RCIC EQUIP ROOM	B21-XY-5949A B21-XY-5949B CH. A1-3	E51-TE-N023A E51-TE-N023B	165	295	5
HPCI	HPCI EQUIP ROOM	B21-XY-5948A B21-XY-5948B CH. A1-1	E41-TE-N030A E41-TE-N030B	165		4
STEAM TUNNEL	RCIC STM TUNNEL	B21-XY-5949A B21-XY-5949B CH. A3-3	E51-TE-N025A E51-TE-N025B	190	295	5
	HPCI STM TUNNEL	B21-XY-5948A B21-XY-5948B CH. A5-1	E51-TE-N025C E51-TE-N025D	190	295	4
20 FT	20 FT NORTH	B21-XY-5948A CH. A1-4	B21-TE-5761A	140	200	N/A
	20 FT SOUTH	B21-XY-5948B CH. A1-4	B21-TE-5763B			
50 FT	50 FT NW	B21-XY-5948A CH. A2-4	B21-TE-5762A	140	200	N/A
	50 FT SE	B21-XY-5948B CH. A2-4	B21-TE-5764B			
REACTOR BLDG	MULTIPLE AREAS	ANNUNCIATOR PANEL A-02	WINDOW 5-7	ALARM SETPOINT	N/A	3,4, AND/OR 5
REACTOR BLDG	MSIV PIT	ANNUNCIATOR PANEL A-06	WINDOW 6-7	ALARM SETPOINT	N/A	1

NOTE 1 MAX NORM OPERATING VALUE IS THE ANNUNCIATOR/GROUP ISOLATION SETPOINT WHERE APPLICABLE

ATTACHMENT 10 (Cont'd)

FIGURE 21
SECONDARY CONTAINMENT AREA DIFFERENTIAL TEMPERATURE

TABLE 2

AREA DIFFERENTIAL TEMPERATURE LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	STEAM LEAK DETECTION CHANNEL	MAX NORM OPERATING VALUE (°F) (NOTE 1)	AUTO GROUP ISOL
RWCU	RWCU PUMP ROOM A	B21-XY-5949A B21-XY-5949B CH. A4-1	45	3
	RWCU PUMP ROOM B	B21-XY-5949A B21-XY-5949B CH. A5-1		
	RWCU HX ROOM	B21-XY-5949A B21-XY-5949B CH. A6-1		
N RHR	N RHR EQUIP ROOM	B21-XY-5948A CH. A6-4	50	N/A
S RHR	S RHR EQUIP ROOM	B21-XY-5948B CH. A6-4	50	N/A
	RCIC EQUIP ROOM	B21-XY-5949A B21-XY-5949B CH. A2-3	45	5
HPCI	HPCI EQUIP ROOM	B21-XY-5948A B21-XY-5948B CH. A3-1	45	N/A
STEAM TUNNEL	RCIC STM TUNNEL	B21-XY-5949A B21-XY-5949B CH. A4-3	45	5
	HPCI STM TUNNEL	B21-XY-5948A B21-XY-5948B CH. A6-1	45	4
REACTOR BLDG	MULTIPLE AREAS	ANNUNCIATOR A-02 6-7	ALARM SETPOINT	3, 4, AND/OR 5

NOTE 1: MAX NORM OPERATING VALUE IS THE ANNUNCIATOR/GROUP ISOLATION SETPOINT WHERE APPLICABLE

ATTACHMENT 10 (Cont'd)

FIGURE 22
SECONDARY CONTAINMENT AREA RADIATION

TABLE 3
AREA RADIATION LIMITS

PLANT AREA	PLANT LOCATION DESCRIPTION	ARM CHANNEL	MAX NORM OPERATING VALUE (mR/HR)	MAX SAFE OPERATING VALUE (mR/HR)
N CORE SPRAY	N CORE SPRAY ROOM	15	200	* 7000
S CORE SPRAY	S CORE SPRAY ROOM	16	200	* 7000
N RHR	N RHR ROOM	17	200	* 7000
S RHR	S RHR ROOM	18	200	* 3000
HPCI	HPCI ROOM	N/A	N/A	* 3000
RX BLDG 20 FT ELEV	N ACROSS FROM TIP ROOM	19	80	* 2000
	DRYWELL ENTRANCE	20		
	DECON ROOM	22		
	RAILROAD DOORS	23		
RX BLDG 50 FT ELEV	SAMPLE STATION	24	80	* 2000
	RX BLDG AIR LOCK	25		
RX BLDG 117 FT ELEV	N OF FUEL STORAGE POOL	27	80	* 7000
	BETWEEN RX & FUEL POOL	28	1000	7000
	CASK WASH AREA	29	90	* 7000
RX BLDG 80 FT ELEV	SPENT FUEL COOLING SYSTEM	30	90	* 3000

* CONTACT E&RC TO DETERMINE IF MAX SAFE OPERATING VALUE IS EXCEEDED

DATE COMPLETED ____________________
UNIT________ % PWR_____ GMWE________
SUPERVISOR____________________
REASON FOR TEST (check one or more):
_____ Routine surveillance
_____ OWP #____________________
_____ WR/JO #____________________
_____ Other (explain)____________________

FREQUENCY:

A. Post Refueling Outage

B. As determined by Reactor Engineering

PLANT OPERATING MANUAL

VOLUME X

PERIODIC TEST

UNIT
0

OPT-50.2

SRM SIGNAL-TO-NOISE RATIO AND IRM RANGE 6/7 OVERLAP DETERMINATION

REVISION 11

EFFECTIVE DATE
22
10-~~15~~-96
DC 10-22-96

Sponsor DM Alford Date 9/27/96

Approval [signature] Manager - Engineering Date 10/10/96

1.0 PURPOSE

The purpose of this procedure is to demonstrate that adequate SRM Signal-To-Noise Ratio exist and to verify correct overlap of the IRM Ranges 6 and 7.

2.0 REFERENCES

2.1 Startup Test Instruction 6 - SRM Performance and Control Rod Sequence, Document No. 22A2229 AJ.

2.2 Startup Test Instruction 10-IRM Performance, Document No. 22A2229 AN.

2.3 OPT-14.3.1, In-Sequence Critical Shutdown Margin Calculation

2.4 OFH-11, Refueling

2.5 OPT-50.1, LPRM and APRM Initial Sensitivities

3.0 PREREQUISITES

The prerequisites for this PT are listed with each section.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The SRM and IRM shorting links are removed to activate the SRM noncoincident scram until OPT-14.3.1, In-Sequence Critical Shutdown Margin Calculation, has been satisfactorily completed. Therefore, control rods should be inserted in reverse order when the power level of 10^5 cps is reached to prevent a scram at 5×10^5 cps prior to installing the shorting links.

4.2 Caution should be used when SRM shorting links are removed or installed so that inadvertent scrams are avoided.

4.3 Independent verification is required in this PT.

5.0 SPECIAL TOOLS AND EQUIPMENT

Electronic calculator

6.0 ACCEPTANCE CRITERIA

6.1 Level 1:

6.1.1 SRMs have a count rate of 3 cps or more when fully inserted in the core.

6.1.2 The signal to noise ratio for the SRMs is two or more.

6.2 Level 2:

6.2.1 IRM readings in Range 7 are within 8 to 12 percent of the readings in Range 6.

7.0 ANNUNCIATIONS EXPECTED

None

8.0 PROCEDURE Initials

8.1 SRM Initial Data Check

NOTE: This section should be performed within one week prior to initial criticality.

8.1.1 Prerequisites

8.1.1.1 OFH-11 fuel loading is complete and all control rods are fully inserted. ________

8.1.1.2 SRMs are operable and fully inserted in the core. ________

8.2 Procedure

8.2.1 Record date and time.

Date __________ Time __________

	Counts Fully Inserted (A)	Counts Fully Withdrawn (B)	Signal to Noise Ratio (A-B)/B
SRM A			
SRM B			
SRM C			
SRM D			

8.2.2 Record the count rate for SRM A when fully inserted in the core. ________

8.2.3 Verify that SRM A has a count rate of ≥ 3 cps when fully inserted. ________

8.2.4 Move SRM A to its fully withdrawn position and record the fully withdrawn count rate. ________

8.2.5 Return SRM A to its fully inserted position. ________

8.2.6 Calculate and record the signal to noise ratio for SRM A as shown. ________

8.2.7 Verify that SRM A has a signal to noise ratio ≥ 2. ________

8.2.8 Repeat Steps 8.2.2 through 8.2.7 for SRM B. ________

8.0 PROCEDURE

Initials

8.2.9 Repeat Steps 8.2.2 through 8.2.7 for SRM C. ________

8.2.10 Repeat Steps 8.2.2 through 8.2.7 for SRM D. ________

8.2.11 Independently verify Signal-To-Noise Ratio calculations in the above table. ____/____ Ind.Ver.

8.2.12 Independently verify SRMs have a count rate ≥ 3 cps when fully inserted into the core. ____/____ Ind.Ver.

8.2.13 Inform the Unit SRO that the Signal-To-Noise Ratio portion of the OPT-50.2 is complete and Sat/Unsat. (circle) ________

8.3 IRM Ranges 6 and 7 Continuity Check

8.3.1 Prerequisites

8.3.1.1 Reactor is critical in the Intermediate (IRM) Range. ________

8.3.2 Procedure Steps

8.3.2.1 Record date and time.

Date __________ Time __________

8.3.2.2 For each IRM, establish a power level high in IRM Range 6. Record reading below. (IRM Range 6 uses the black 0-125 scale.) ________

8.3.2.3 For each IRM, switch the IRM to Range 7 and record reading. (IRM Range 7 uses the red 0-40 scale.) Calculate percentage of Range 7 to Range 6 readings by the formula: ________

Percent = ((Range 7 / Range 6) * 100)

IRMs

	A	B	C	D	E	F	G	H
Range 6								
Range 7								
Percent								

8.3.2.4 Verify that the IRM readings in Range 7 are within 8 to 12 percent of the readings in Range 6. ________

8.0 PROCEDURE

Initials

8.3.2.5 Independently verify calculations in Step 8.3.2.3 and that readings in Range 7 are within 8 to 12 percent of readings in Range 6. If readings are outside the 8 to 12 percent band, a calibration of the IRM amplifiers should be performed. ____/____ Ind.Ver.

8.3.2.6 If adjustments were required per Step 8.3.2.5, record data following the final adjustment, for the applicable IRM(s) in the table below. ________

IRM	A	B	C	D	E	F	G	H
Range 6								
Range 7								
Percent								

8.3.2.7 Independently verify calculations performed in Step 8.3.2.6 and that readings in Range 7 are within 8 to 12 percent of readings in Range 6. ____/____ Ind.Ver.

8.4 Inform Unit SRO of PT status (SAT/UNSAT). ________

8.5 Inform the Neutron System Engineer if calibrations were required (Yes/No) and of the PT status (Sat/Unsat). ________

9.0 General Comments/Recommendation ______________________________

____

____

____

____

____

____

____

____

	Initials	Name (Print)
PT performed by	________	______________________________
	________	______________________________
	________	______________________________
	________	______________________________

PT has been satisfactorily completed

Responsible Engineering Manager ______________________________
(Signature)

PT has NOT been satisfactorily completed

Responsible Engineering Manager ______________________________
(Signature)

Reviewed by ______________________________
(Signature)

Corrective Action Required: a. ( ) None
b. ( ) WR/JO No. ____________
c. ( ) Other (explain)

I
Information
Use

Data Sheet for Battery Ground Detection

BATTERY (1A)
DATE: TIME: PERFORMED BY:
CURRENT: "P" mA "PN" mA "N" mA
BATTERY BUS VOLTAGE: "1A-1" V DC "1A-2" V DC
"1A" RESISTANCE = $\dfrac{\text{V DC + V DC}}{\text{(P)mA + (N)mA}}$ - 50 = $\boxed{}$
BATTERY (1B)
DATE: TIME: PERFORMED BY:
CURRENT: "P" mA "PN" mA "N" mA
BATTERY BUS VOLTAGE: "1B-1" V DC "1B-2" V DC
"1B" RESISTANCE = $\dfrac{\text{V DC + V DC}}{\text{(P)mA + (N)mA}}$ - 50 = $\boxed{}$

EXAMPLE:

BATTERY (1B)
DATE: 6/5/91 TIME: 1302 PERFORMED BY: XXXXXXXXXXX
CURRENT: "P" 2.15 mA "PN" 0.08 mA "N" 1.75 mA
BATTERY BUS VOLTAGE: "1B-1" 135 V DC "1B-2" 140 V DC
"1B" RESISTANCE = $\dfrac{\text{135V DC + 140V DC}}{\text{2.15 mA + 1.75 mA}}$ - 50 = $\boxed{}$

The "- 50" factor in the equation is to account for the presence of a 50 KΩ resistor in series with the milliamp meter.

Since the overall resistance (20.5 KΩ) is below the setpoint (25 KΩ), this would suggest there is a ground on the system which will first need to be verified. Once the ground has been verified, ground hunting activities should commence. According to Attachment 2, the ground is located on the N Bus. AI-115 should then be referenced for monitoring activities which will show that Action Level 1 should be entered.

CP&L

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK NUCLEAR PLANT

I
Information Use

PLANT OPERATING MANUAL

VOLUME I
BOOK 2

ADMINISTRATIVE INSTRUCTION

DISTRIBUTION
SCOPE

UNIT
0

0AI-115

125/250 VDC SYSTEM GROUND CORRECTION GUIDELINES

REVISION 2

RECEIVED BY BNP
JUN 0 8 1995
NUCLEAR DOCUMENT CONTROL

EFFECTIVE DATE
6-8-95

Sponsor Jeff Cannon Date 5-9-95

Approval Plant General Manager (U1/U2)/ Manager - Technical Support Date 6/7/95

LIST OF EFFECTIVE PAGES

OAI-115

Page(s)	Revision
1-2	2
3	0
4-8	2

1.0 DC GROUND POLICY STATEMENT

It is the policy of Carolina Power & Light Company to engineer, construct, and operate nuclear power plants without jeopardy to the health and safety of the public and of its employees.

The function of the this procedure is to establish a set of criteria for actions which should be taken based upon the magnitude of grounds on the 125/250 VDC distribution system. Completion of the various actions will provide a reasonable assurance that the 125/250 VDC distribution system will continue to satisfy the CP&L Policy.

It shall be the responsibility of all organizations at the plant and of all corporate support groups to provide support needed to ensure that the DC Ground program is successful.

2.0 PURPOSE AND SCOPE

The 125/250 VDC system at BNP is an ungrounded system. The system was designed as an ungrounded system so that the system could sustain one solid ground fault and still be able to meet all requirements for continued safe operation. On the ungrounded system when the first ground fault occurs the system is degraded but it is not degraded below an acceptable grounded level.

The dc ground detectors provide a means of monitoring the dc distribution system for significant grounds. If one ground occurs the ground detectors will alert plant operations so that the ground can be corrected before a second more significant ground occurs.

The purpose of this document is to provide a set of guidelines for plant operations to use to determine the urgency of action and the actions to be taken to correct a known dc ground based upon the severity of the ground. The initial values selected for this document should provide a reasonable assurance that the operation of the dc distribution system will not be adversely affected.

The effectiveness of this document will depend on the full support of all levels of plant and corporate management. This includes the provision of adequate resources and a dedication to correcting grounds expeditiously as required by the action statements.

Changes to these guidelines will be made as the detailed evaluation of the dc distribution system continues and as refinements and improvements in the dc ground detection system dictate.

3.0 BASIS FOR ACTION LEVELS

Electrical Evaluation BNP-E-6.116 was issued to provide a basis for the setpoint of the battery ground detectors. The value established by this evaluation is 25K ohms. The ground detectors operate with a ± 15% band, which corresponds to 21.3K to 28.8K ohms.

Electrical devices at BNP were researched and the most sensitive device of concern was determined to be GE HFA relays. These relays have a nominal dropout current of 3.75 ma. This value was utilized to derive an appropriate setpoint. Therefore, at ground levels just below 25K ohms, situations (involving <u>two</u> very strategically located grounds) could develop that might hold-in a normally energized relay.

At ground levels below 15K ohms, sufficient currents are allowed to develop which under the right conditions (involving <u>two</u> strategically located grounds) could result in picking up a de-energized relay. Since most relays in the system are normally de-energized relays, this level is considered the most urgent to correct.

Above the ground detector setpoint of 25K ohms, there are no required actions to be taken. Action Level 1 is between 25K and 15K ohms and plant procedures should be entered to locate and correct the ground condition. Grounds below 15K ohms shall be considered the most severe condition (Action Level 2) and shall require the greatest degree of effort towards resolution.

This procedure provides guidance to plant operations as to prudent actions to be taken when DC System grounds are encountered. These guidelines do not necessarily ensure that a system operating problem will not occur if the guidelines are followed. However, due to the fact that very specific, isolated, and independent events must occur with the DC grounds on the system to create the potential for operational problems, the guidelines provide a relative degree of confidence that corrective actions can be completed before these system operating problems occur.

4.0 RESPONSIBILITIES

4.1 Plant General Manager is responsible for establishing and maintaining the dc ground program and shall assign specific responsibilities for the implementation of the program.

4.2 The Manager of Operations is responsible for ensuring that appropriate corrective actions are followed as outlined in this guideline.

4.3 The Manager of Maintenance is responsible for assuring prompt and effective repair of plant equipment with dc grounds.

4.4 The Manager of Technical Support is responsible for any support functions towards monitoring and/or repair of dc grounds.

5.0 DEFINITIONS

5.1 Action Levels

Operation at or below the action-level 1 or 2 values presented in this document may not result in immediate misoperation of dc electrical equipment, however, the dc electrical equipment will be operating in a degraded condition.

5.1.1 Action Level 0 (Ground Resistance ≥ 25kΩ)

This is the achievable value which should be maintained by applying good operating and maintenance practices. Every effort should be made to operate with the dc grounds at or above this value.

5.1.2 Action Level 1 (Ground Resistance 15kΩ to 25kΩ)

This level represents the level of operation of the dc system below the ground detector alarm setpoint of 25kΩ. Operation with one ground in this range will not result in the inadvertent pickup of a deenergized relay, but could result in the inadvertent hold-in of an energized relay if a second hard ground were to occur. Prompt actions should be taken to restore the ground resistance to Action Level 0 values to minimize the possibility of continued degradation of the dc distribution system.

5.1.3 Action Level 2 (Ground Resistance ≤ 15kΩ)

This level represents the level of operation that could result in the inadvertent pickup of a deenergized relay if a second ground ≤25KΩ occurs. Immediate actions shall be taken to restore the ground resistance to a minimum of Action Level 1 values prior to encountering a second ground ≤25KΩ. Continuous investigation of possible grounds should be conducted until the ground is located and corrected.

6.0 ACTION REQUIREMENTS

6.1 Action Level 0 (≥25kΩ)

At Action Level 0 the dc grounds are above the dc ground detector alarm point of 25kΩ. No actions are required providing the dc ground detector alarms are operable.

If the dc ground detector alarms are determined to be inoperable, then the ground current shall be measured once per shift per OP-51, to verify that the grounds are above the alarm point.

6.1.1 If the DC ground annunciator alarms or ground current measurements demonstrate a ground <25kΩ, perform the following:

6.1.1.1 Note ground in Shift Supervisor's log along with any possibly related activity/evolution.

6.1.1.2 If ground measurement determines that ground is actually below 25kΩ, Operations should perform ground hunting per OP-51 and appropriate action level.

6.2 Action Level 1 (<25kΩ to >15kΩ)

At Action Level 1 the dc grounds are above the point where deenergized relays may pickup if a hard ground occurs on the opposite bus. This action level shall be entered if ground current measurements demonstrate that a ground between 25kΩ and 15kΩ actually exists. Action steps for Action Level 1 are as follows:

6.2.1 The ground current shall be measured once per shift per OP-51, to ensure that the ground resistance has not degraded to the Action Level 2 point.

6.2.2 Efforts shall be undertaken to isolate all circuits per OP-51 which can be isolated without affecting plant operation in an effort to locate the ground. The ground hunting effort will be conducted on an eight hour per day five days per week basis.

6.0 ACTION REQUIREMENTS (Continued)

6.2 Action Level 1 (<25kΩ to >15kΩ) (Continued)

6.2.3 If the ground cannot be located within 14 days then the actions Steps 6.3.2 through 6.3.4 of Action Level 2 shall be implemented.

6.2.4 Circuits identified with grounds should be corrected expeditiously, and if possible, placed under clearance while waiting correction.

6.3 Action Level 2 (≤15kΩ)

At Action Level 2 the dc grounds are at the point where deenergized relays may pickup or energized relays may dropout if another ≤25kΩ ground occurs on the opposite bus. This action level shall be entered if measurement of the ground current demonstrates that the ground is less than or equal to 15kΩ. Action steps for Action Level 2 are as follows:

6.3.1 The ground current shall be measured twice per shift to ensure that the ground resistance has not degraded or to determine if any worse grounds develop.

6.3.2 Efforts shall be immediately undertaken to isolate all dc circuits per OP-51 which have not been isolated. The ground hunting effort will be conducted on an 24 hour per day seven day per week basis.

6.3.3 If after seven days the ground has not been located and no more circuits can be isolated per OP-51 due to the existing plant operating conditions, PNSC is to approve subsequent ground hunting actions.

6.3.4 Circuits identified with grounds should be corrected expeditiously, and if possible, placed under clearance while waiting correction.

7.0 REFERENCES

7.1 FP-84882, Battery Ground Detector

7.2 IEN 88-86, Operation with Multiple Grounds in Direct Current Distribution Systems, October 21, 1988

7.3 IEN 88-86, Supplement 1, Operation with Multiple Grounds in Direct Current Distribution Systems, March 31, 1989

7.0 REFERENCES (Continued)

7.4 SDCD-51, System Design Criteria Document for the DC Electrical System, April 9, 1988

7.5 Commonwealth Edison Company, Ground Task Force Final Report, May 25, 1989

7.6 Calculation BNP-E-6.116, Ground Detection Setpoint Basis for 125/250V DC System

ATTACHMENT L (Cont'd)

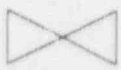
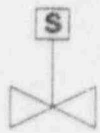
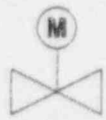
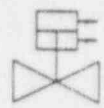
Table 1
Tagout Devices

DISCUSSION

Table 1 lists various devices that may be used as isolation boundaries for clearances. The suggested tagout method and known restrictions on use for clearances are given. If it is desired to use a particular device and actuator not shown, or to use a restricted device shown, a special evaluation must be performed by a person knowledgeable of the device, and the evaluation logged or otherwise documented.

Clearance Boundary Isolation Components

TAGOUT DEVICE: GATE VALVE

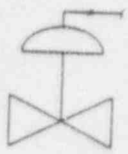
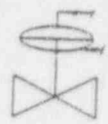
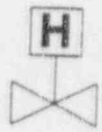
	MANUAL	SOLENOID	MOTOR	CYLINDER
				
TAGOUT METHOD	Tag on manual operator.		Tag on motor breaker ("BKR Locked Off or Removed"). Tag on manual operator.	Tag on air supply isolation ("Closed"). Tag on gag device ("Gagged Closed"). Tag on actuator ("Do not remove actuator").
RESTRICTIONS	Consider leakage history and use two valves in series if leakage unacceptable.	Do not use as tagout boundary if not listed on drawing as a fail-closed valve.		If valve is marked fail-closed on print, no gag is needed, unless system pressure is expected to exceed valve spring pressure.

ATTACHMENT L (Cont'd)

Table 1
Tagout Devices

Clearance Boundary Isolation Components

<u>TAGOUT DEVICE</u>: GATE VALVE

	SPRING-OPPOSED PNEUMATIC DIAPHRAGM	PRESSURE-BALANCED DIAPHRAGM	HYDRAULIC	
				
TAGOUT METHOD	Tag on air supply isolation ("Closed"). Tag on gag device ("Gagged Closed").			
RESTRICTIONS	Must use gag device. Prone to leakage. Consider leakage history & use two valve isolation if necessary.	DO NOT USE AS CLEARANCE BOUNDARY.	Use as clearance boundary is dependent on valve model. spring closure. and hydraulic supply. Evaluate case-by-case for use.	

ATTACHMENT L (Cont'd)

Table 1
Tagout Devices

<u>TAGOUT DEVICE</u>: GLOBE VALVE

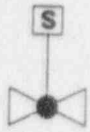
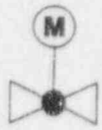
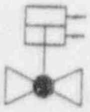
	MANUAL	SOLENOID	MOTOR	CYLINDER
				
TAGOUT METHOD	Tag on manual operator.	Tag on solenoid power supply breaker. NOTE: Other components may receive feed from same breaker.	Tag on motor breaker ("BKR Locked Off or Removed"). Tag on manual operator.	Tag on air supply isolation ("Closed"). Tag on gag device ("Gagged Closed"). Gagging device needed.
RESTRICTIONS		Use as tagout boundary only when marked on drawings as F.C.	Do not use as isolation boundary if valve is normally used as a flow control valve.	Do not use as isolation boundary if valve is normally used as a flow control valve. Gagging device needed.

Table 1
Tagout Devices

TAGOUT DEVICE: GLOBE VALVE

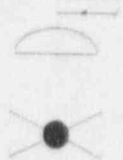
	SPRING-OPPOSED PNEUMATIC DIAPHRAGM			
				
TAGOUT METHOD	Tag on air supply isolation ("Closed"). Tag on gag device ("Gagged Closed"). Gagging device needed.			
RESTRICTIONS	Do not use as isolation boundary if valve is normally used as a flow control valve. Gagging device needed.			

Table 1
Tagout Devices

TAGOUT DEVICE: BUTTERFLY VALVE

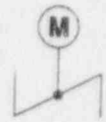
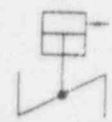
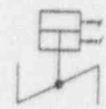
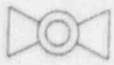
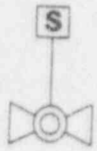
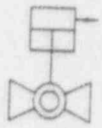
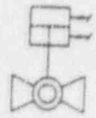
	MANUAL	MOTOR	CYLINDER (SINGLE-ACTING)	CYLINDER (DOUBLE ACTING)
				
TAGOUT METHOD	Tag manual operator.	Tag motor breaker ("BKR Locked Off or Removed"). Tag manual operator.	Tag air supply isolation valve ("Closed"). Tag gag device ("Gagged Closed").	Tag air supply isolation valve ("Closed"). Tag gag device ("Gagged Closed").
RESTRICTIONS	Valves are prone to leak by. Consider leakage history. Use redundant valves if leakage unacceptable. Some valves can leak by due to being closed past the seat.	Valves are prone to leak by. Consider leakage history. Use redundant valves if leakage unacceptable. Some valves can leak by due to being closed past the seat.	Valves are prone to leak by. Consider leakage history. Use redundant valves if leakage unacceptable. Some valves can leak by due to being closed past the seat. Gagging device needed if positioning opposite of fail position.	Valves are prone to leak by. Consider leakage history. Use redundant valves if leakage unaccept-able. Some valves can leak by due to being closed past the seat. Gagging device needed.

Table 1
Tagout Devices

TAGOUT DEVICE: BALL VALVE

	MANUAL	SOLENOID	CYLINDER (SINGLE-ACTING)	CYLINDER (DOUBLE ACTING)
				
TAGOUT METHOD	Tag manual operator.		Tag on air supply isolation valve ("Closed"). Tag on gag device ("Gagged Closed").	Tag on air supply isolation valve ("Closed"). Tag on gag device ("Gagged Closed").
RESTRICTIONS		DO NOT USE AS TAGOUT BOUNDARY.	If valve is marked fail-closed on print, no gag is needed.	Gagging device needed.

ATTACHMENT L (Cont'd)

Table 1
Tagout Devices

<u>TAGOUT DEVICE</u>: PLUG VALVE

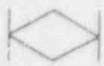
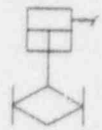
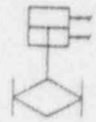
	MANUAL		CYLINDER (SINGLE-ACTING)	CYLINDER (DOUBLE ACTING)
				
TAGOUT METHOD	Tag manual operator.		Tag on air supply isolation valve ("Closed"). Tag on gag device.	Tag on air supply isolation valve ("Closed"). Tag on gag device.
RESTRICTIONS	Prone to leak by. Consider valve history and apply redundant isolation if leakage not acceptable.		Prone to leak by. Consider valve history and apply redundant isolation if leakage not acceptable. No gagging device needed in fail position.	Prone to leak by. Consider valve history and apply redundant isolation if leakage not acceptable. Position valve in desired position prior to isolating air.

Table 1
Tagout Devices

TAGOUT DEVICE: ANGLE VALVE

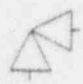
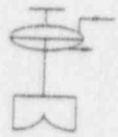
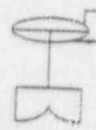
	MANUAL	MOTOR		
				
TAGOUT METHOD	Tag on manual operator.	Tag on motor breaker ("BKR Locked Off or Removed"). Tag on manual operator.		
RESTRICTIONS				

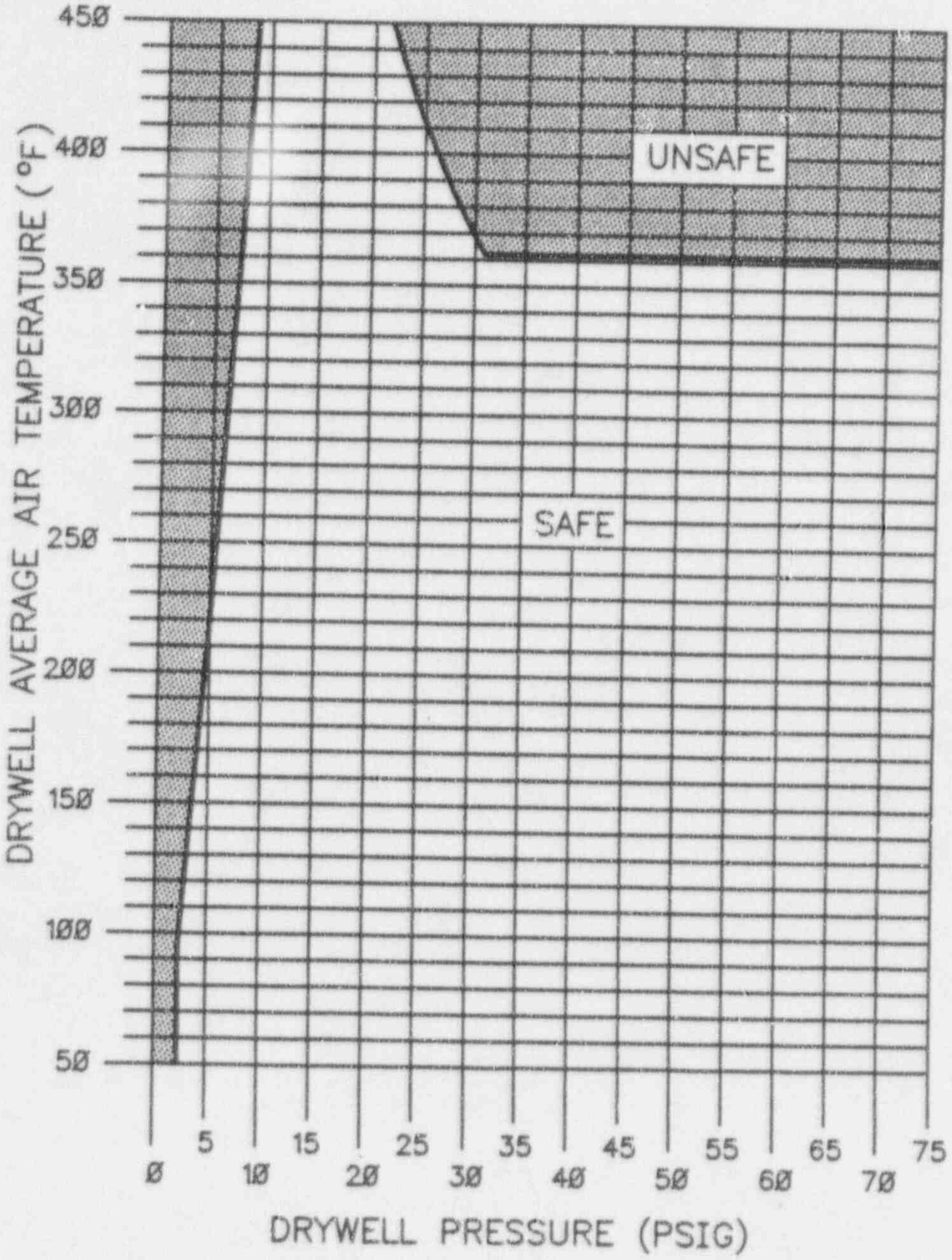
Table 1
Tagout Devices

<u>TAGOUT DEVICE</u>: DIAPHRAGM (PINCH) VALVE

	MANUAL	PRESSURE BALANCED DIAPHRAGM (WITH HANDWHEEL)	PRESSURE BALANCED DIAPHRAGM (NO HANDWHEEL)	
				
TAGOUT METHOD	Tag on handwheel.	Tag on Handwheel.		
RESTRICTIONS		Record throttle position of valve and restore to this position on return to service.	DO NOT USE AS TAGOUT BOUNDARY.	

ATTACHMENT 5 (Cont'd)

FIGURE 1
DRYWELL SPRAY INITIATION LIMIT



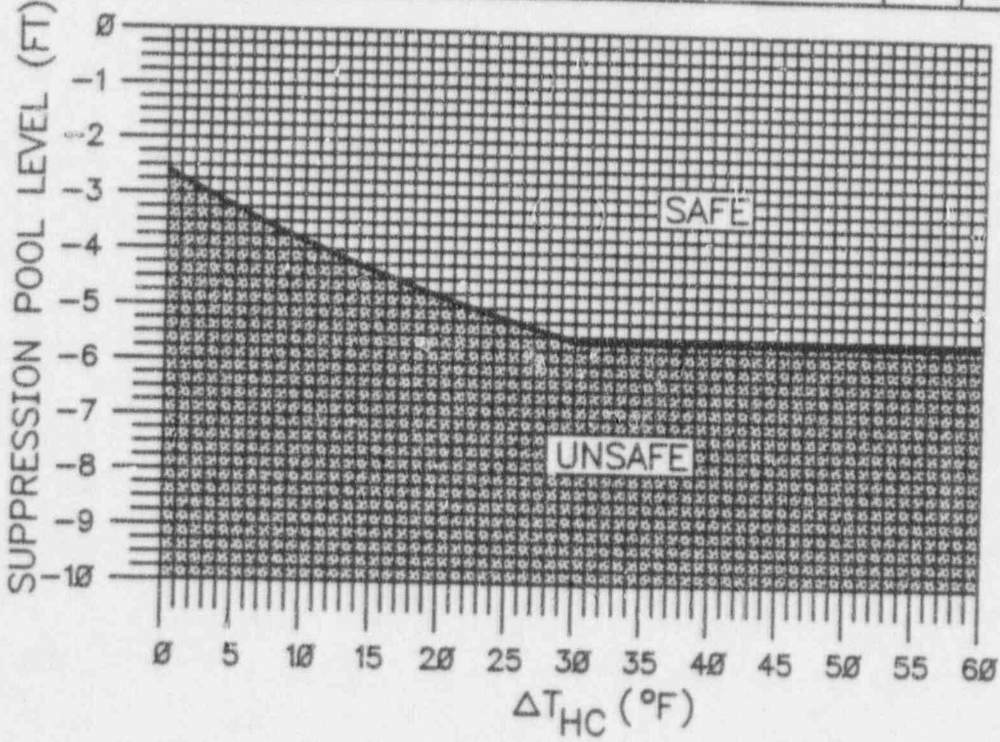
NOTE

DRYWELL AVERAGE AIR TEMPERATURE IS DETERMINED USING ATTACHMENT 4 OF THE "USER'S GUIDE"

ATTACHMENT 5 (Cont'd)

FIGURE 2
UNIT 1 HEAT CAPACITY LEVEL LIMIT

REACTOR PRESSURE	SUPPRESSION POOL TEMPERATURE LIMIT	MINUS	ACTUAL SUPPRESSION POOL WATER TEMPERATURE	EQUALS	ΔT_{HC} (°F)
1000 TO 1130 PSIG	158°F	–		=	
900 TO 1000 PSIG	163°F	–		=	
800 TO 900 PSIG	167°F	–		=	
700 TO 800 PSIG	171°F	–		=	
600 TO 700 PSIG	175°F	–		=	
500 TO 600 PSIG	180°F	–		=	
400 TO 500 PSIG	184°F	–		=	
300 TO 400 PSIG	189°F	–		=	
200 TO 300 PSIG	195°F	–		=	
100 TO 200 PSIG	202°F	–		=	
50 TO 100 PSIG	212°F	–		=	
LESS THAN 50 PSIG	ΔT_{HC} (°F)				62°F



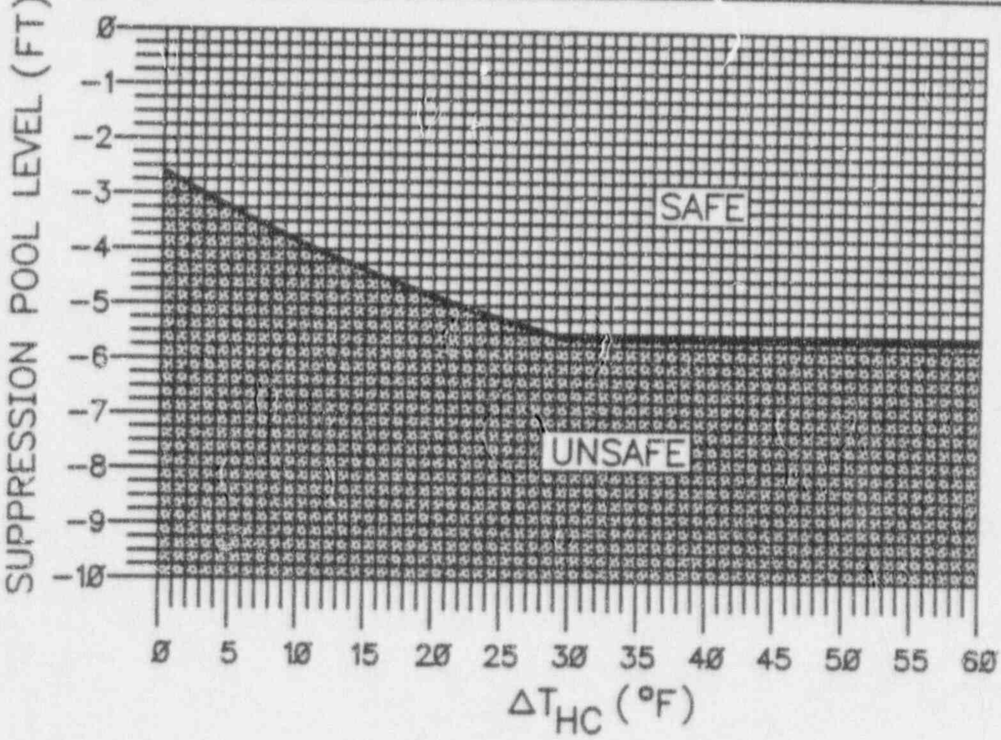
NOTE

SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:
POINT A ON CAC-TR-4426-1A
OR POINT A ON CAC-TR-4426-2A
OR COMPUTER POINT G050
OR COMPUTER POINT G051
OR CAC-TY-4426-1
OR CAC-TY-4426-2

ATTACHMENT 5 (Cont'd)

FIGURE 2A
UNIT 2 HEAT CAPACITY LEVEL LIMIT

REACTOR PRESSURE	SUPPRESSION POOL TEMPERATURE LIMIT	MINUS	ACTUAL SUPPRESSION POOL WATER TEMPERATURE	EQUALS	ΔT_{HC} °F
1ØØØ TO 11Ø5 PSIG	16Ø°F	–		=	
9ØØ TO 1ØØØ PSIG	164°F	–		=	
8ØØ TO 9ØØ PSIG	168°F	–		=	
7ØØ TO 8ØØ PSIG	172°F	–		=	
6ØØ TO 7ØØ PSIG	176°F	–		=	
5ØØ TO 6ØØ PSIG	18Ø°F	–		=	
4ØØ TO 5ØØ PSIG	185°F	–		=	
3ØØ TO 4ØØ PSIG	19Ø°F	–		=	
2ØØ TO 3ØØ PSIG	195°F	–		=	
1ØØ TO 2ØØ PSIG	2Ø2°F	–		=	
5Ø TO 1ØØ PSIG	212°F	–		=	
LESS THAN 5Ø PSIG	ΔT_{HC} °F				6Ø°F

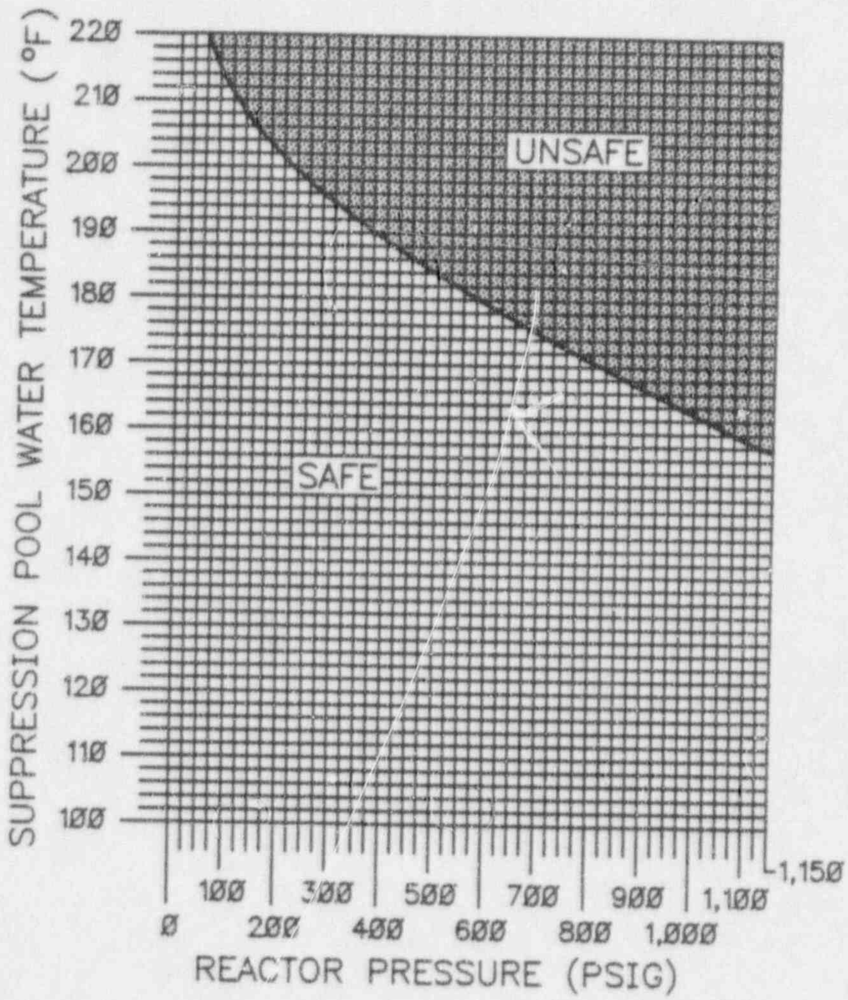


<u>NOTE</u>

SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:
POINT A ON CAC-TR-4426-1A
<u>OR</u> POINT A ON CAC-TR-4426-2A
<u>OR</u> COMPUTER POINT G050
<u>OR</u> COMPUTER POINT G051
<u>OR</u> CAC-TY-4426-1
<u>OR</u> CAC-TY-4426-2

ATTACHMENT 5 (Cont'd)

FIGURE 3
UNIT 1 HEAT CAPACITY TEMPERATURE LIMIT*



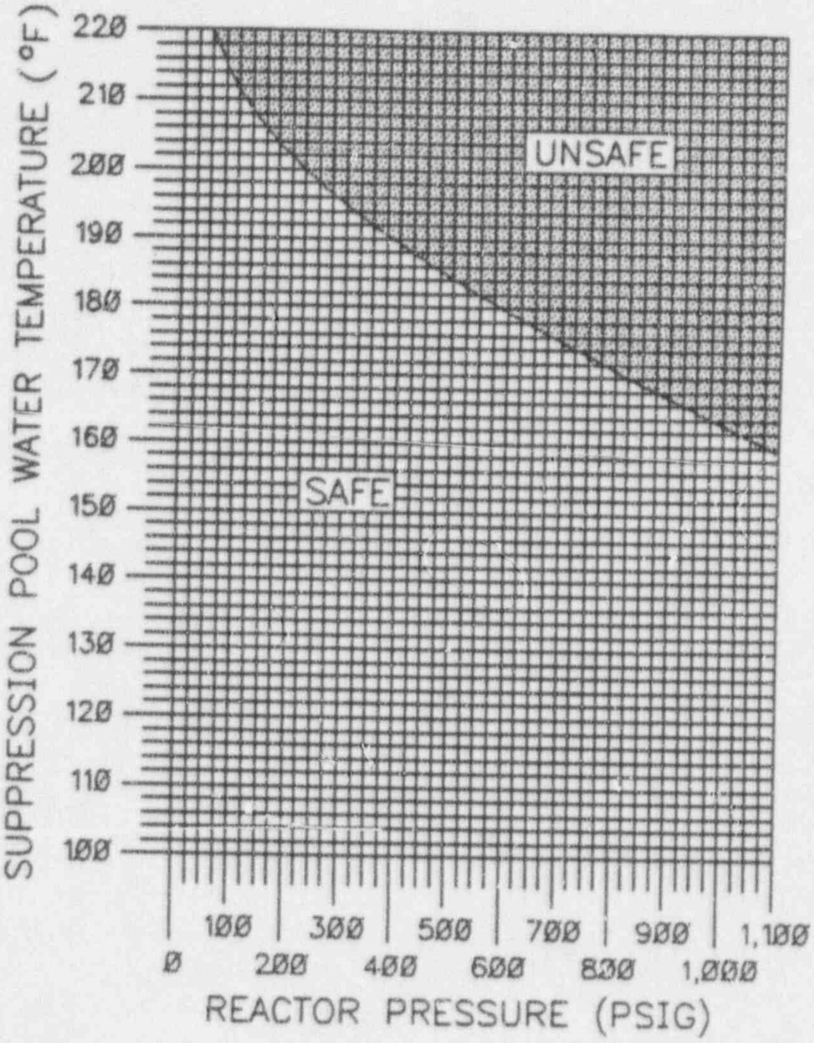
<u>NOTE</u>

SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:
POINT A ON CAC-TR-4426-1A
<u>OR</u> POINT A ON CAC-TR-4426-2A
<u>OR</u> COMPUTER POINT G050
<u>OR</u> COMPUTER POINT G051
<u>OR</u> CAC-TY-4426-1
<u>OR</u> CAC-TY-4426-2

*VALID FOR SUPPRESSION POOL LEVELS ABOVE -31 INCHES

ATTACHMENT 5 (Cont'd)

FIGURE 3A
UNIT 2 HEAT CAPACITY TEMPERATURE LIMIT*



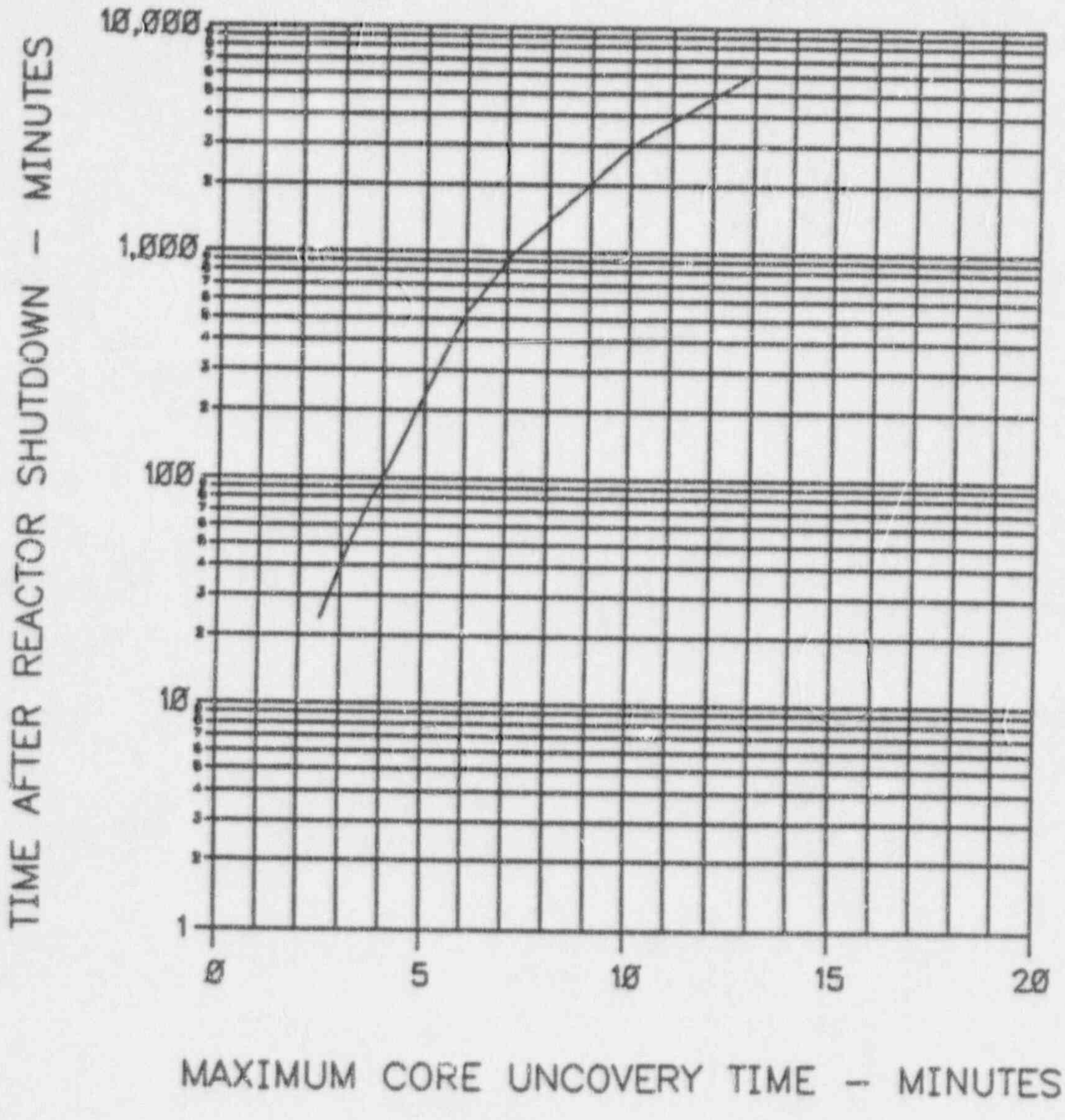
NOTE

SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:
POINT A ON CAC-TR-4426-1A
OR POINT A ON CAC-TR-4426-2A
OR COMPUTER POINT G050
OR COMPUTER POINT G051
OR CAC-TY-4426-1
OR CAC-TY-4426-2

*VALID FOR SUPPRESSION POOL LEVELS ABOVE -31 INCHES

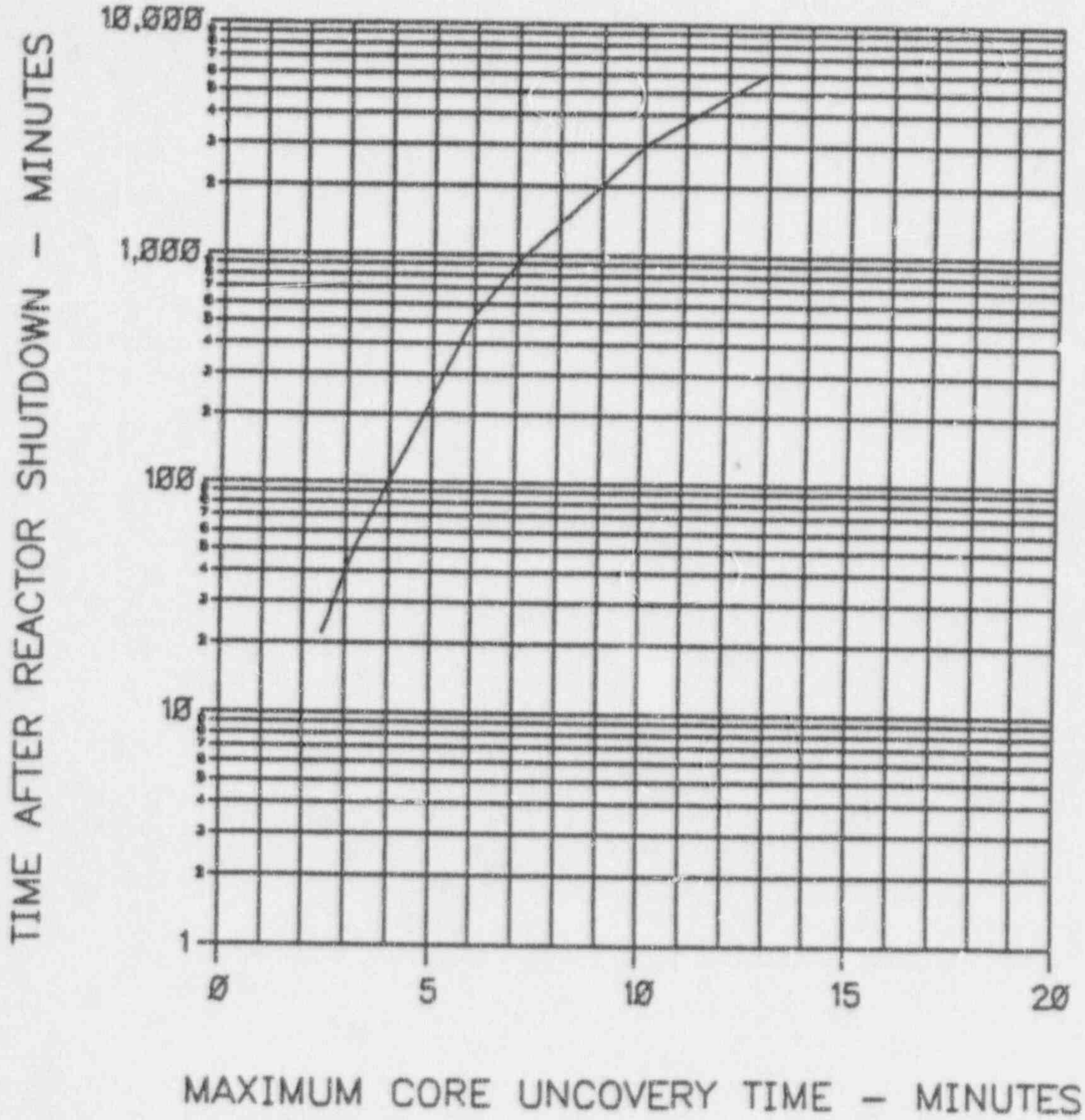
ATTACHMENT 5 (Cont'd)

FIGURE 4
UNIT 1 MAXIMUM CORE UNCOVERY TIME LIMIT



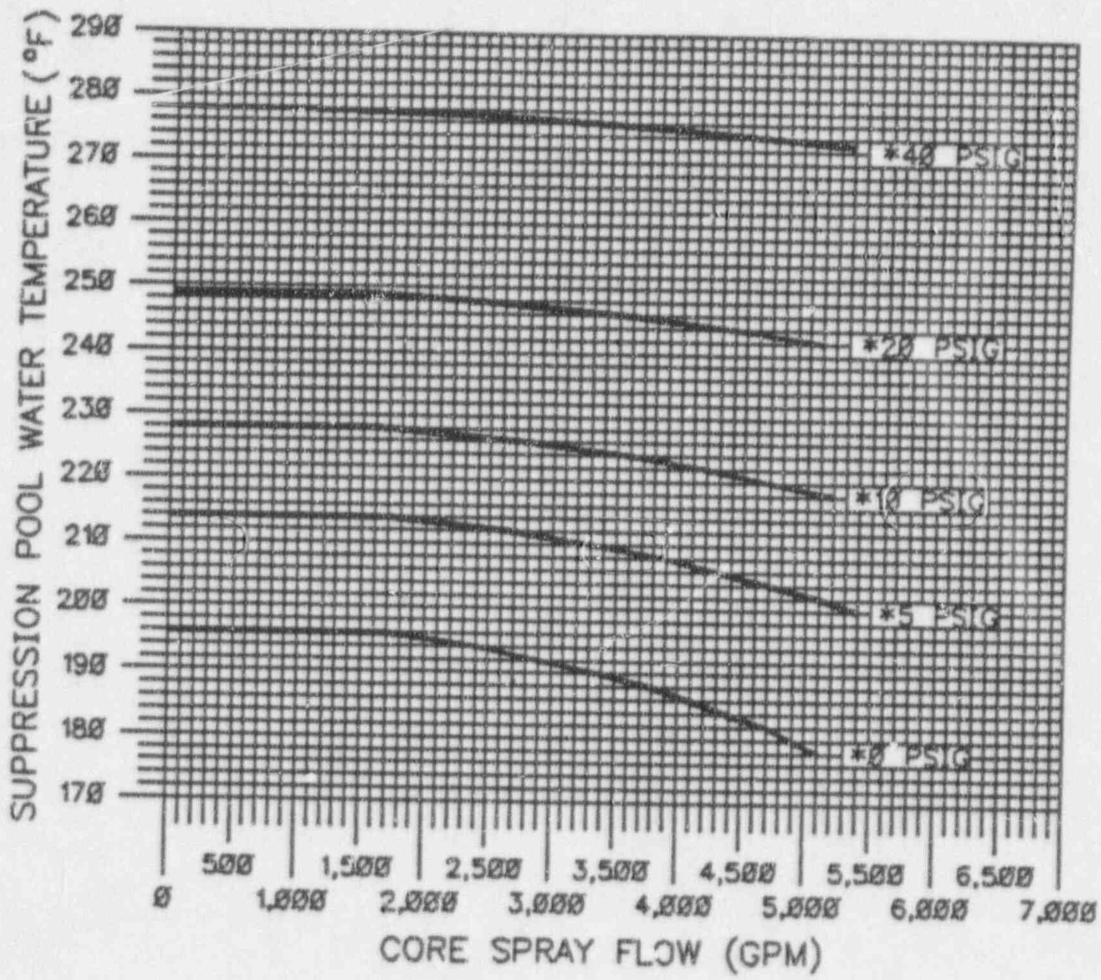
ATTACHMENT 5 (Cont'd)

FIGURE 4A
UNIT 2 MAXIMUM CORE UNCOVERY TIME LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 5
CORE SPRAY NPSH LIMIT

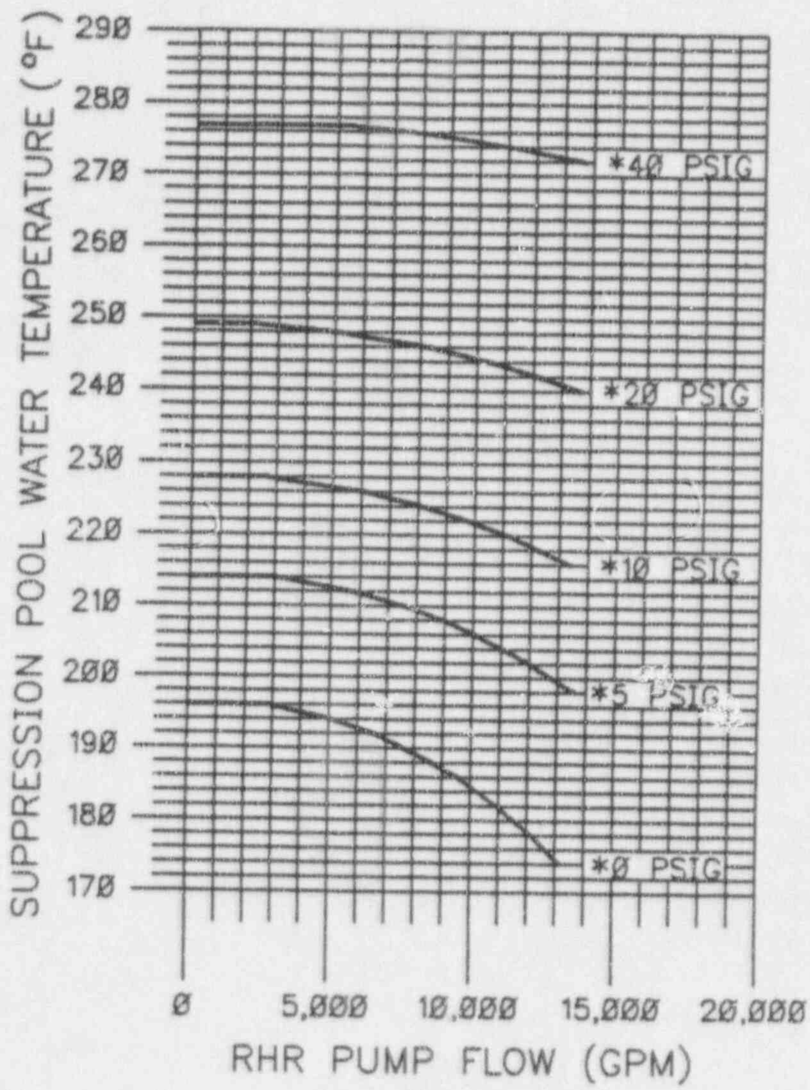


NOTE

SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

ATTACHMENT 5 (Cont'd)

FIGURE 6
RHR NPSH LIMIT

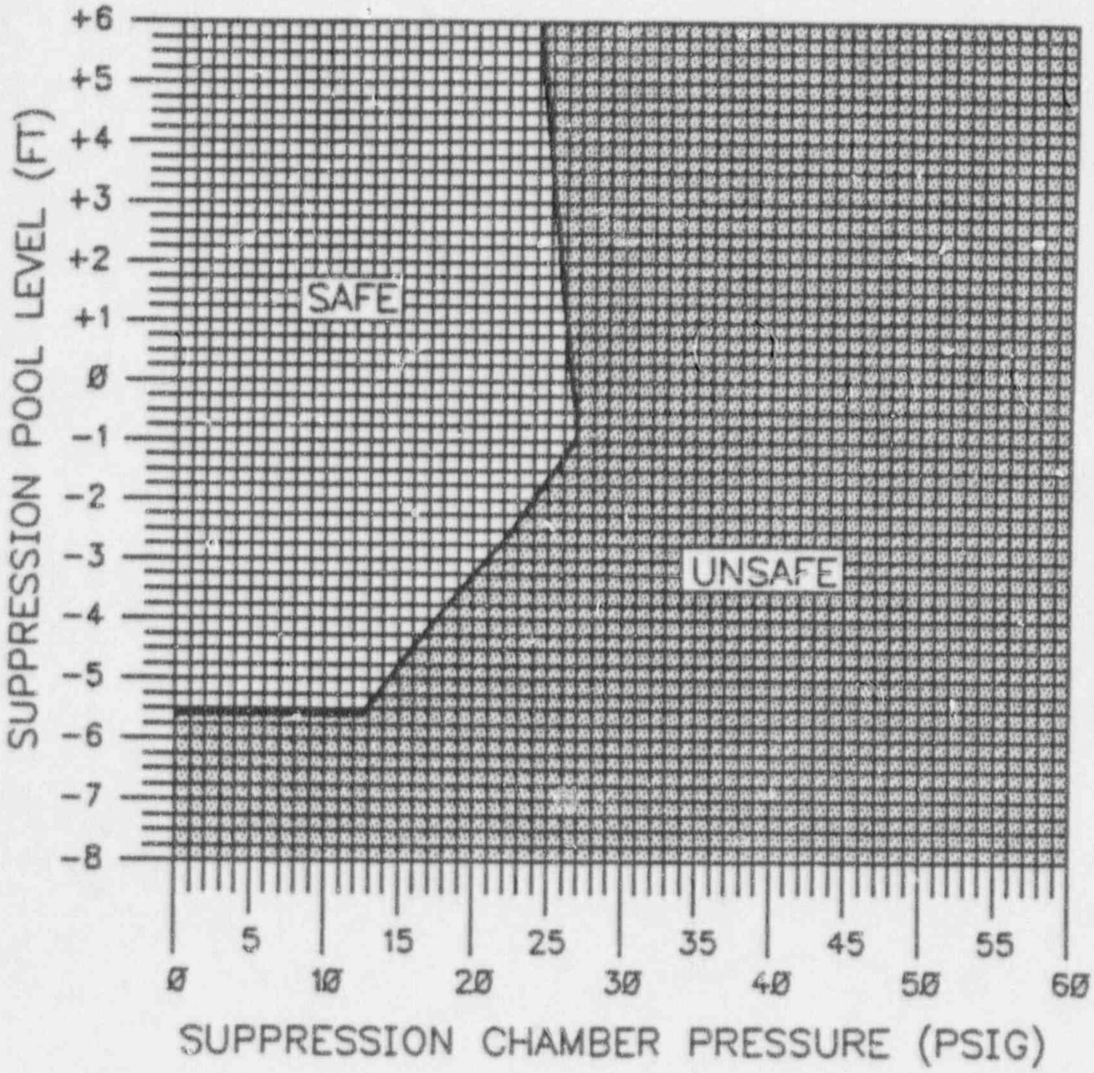


NOTE

SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

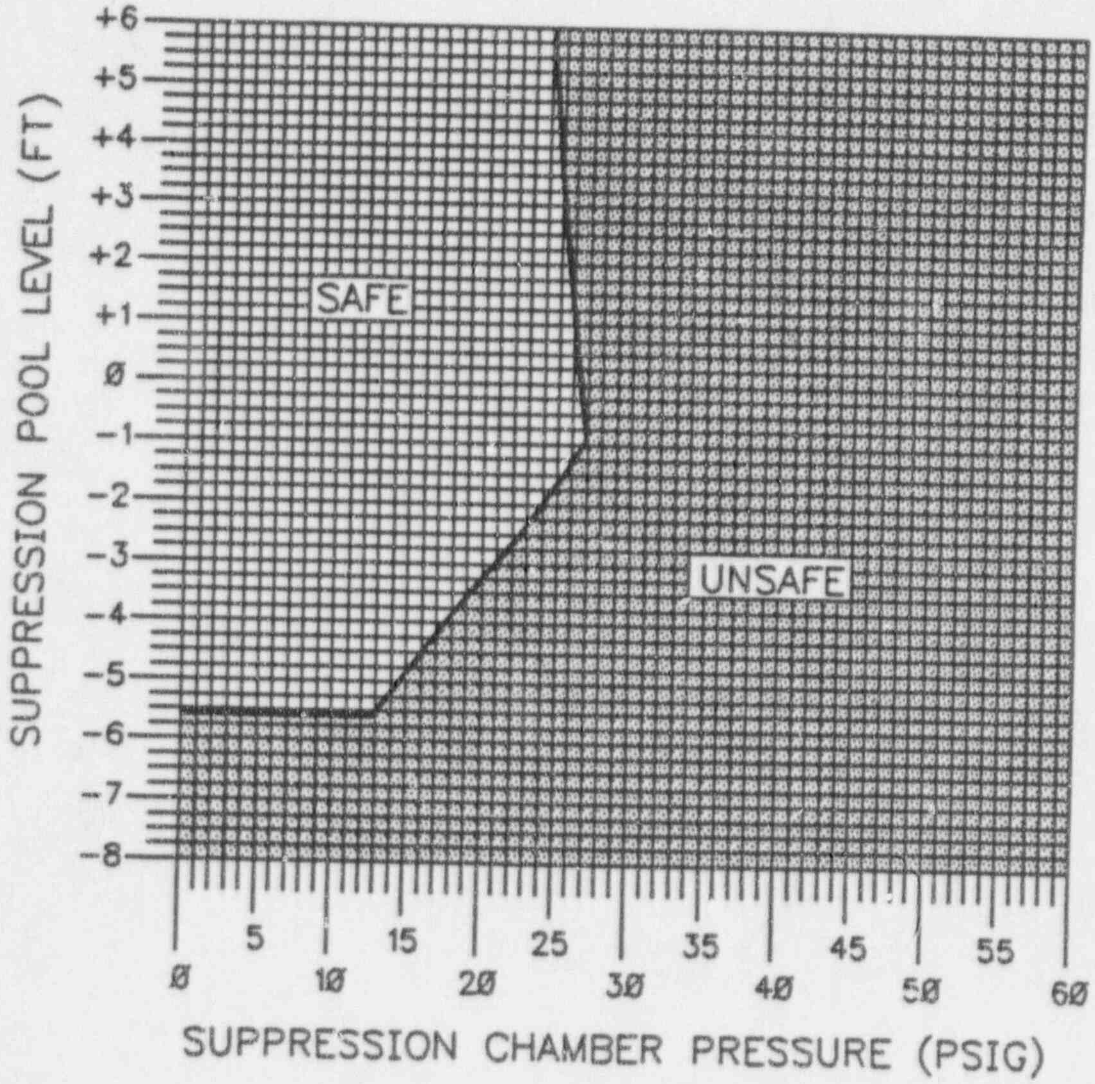
ATTACHMENT 5 (Cont'd)

FIGURE 7
UNIT 1 PRESSURE SUPPRESSION PRESSURE



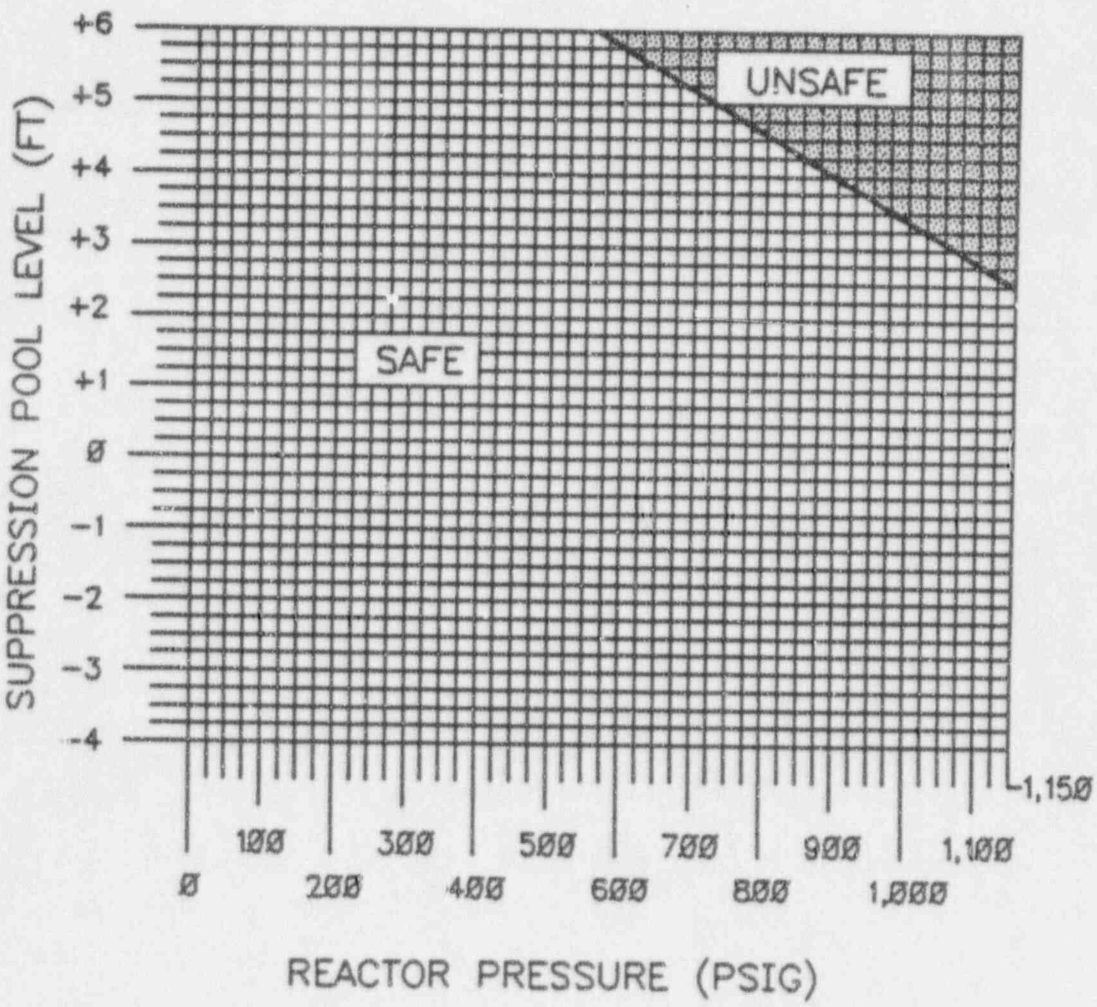
ATTACHMENT 5 (Cont'd)

FIGURE 7A
UNIT 2 PRESSURE SUPPRESSION PRESSURE



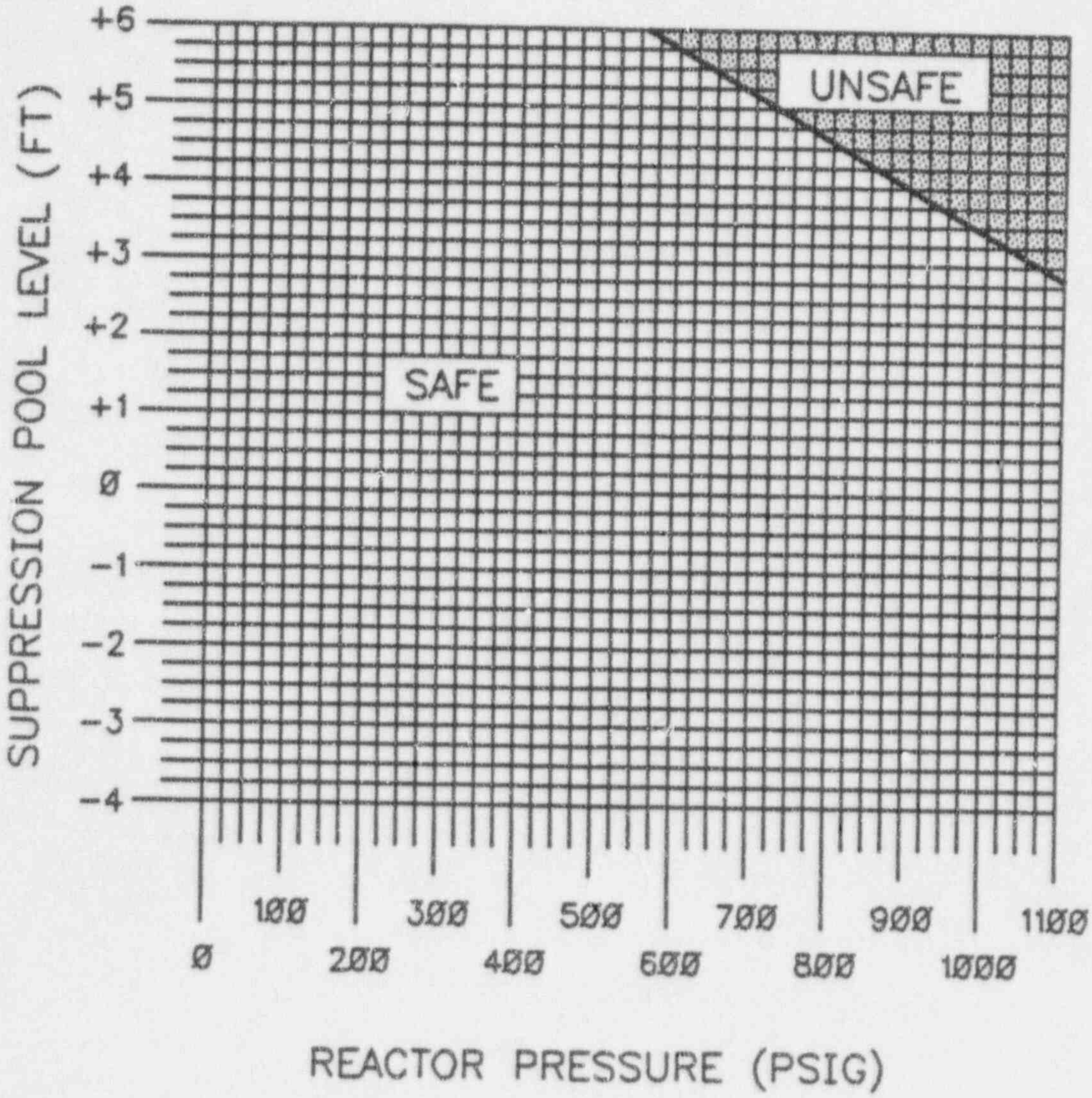
ATTACHMENT 5 (Cont'd)

FIGURE 8
UNIT 1 SRV TAIL PIPE LEVEL LIMIT



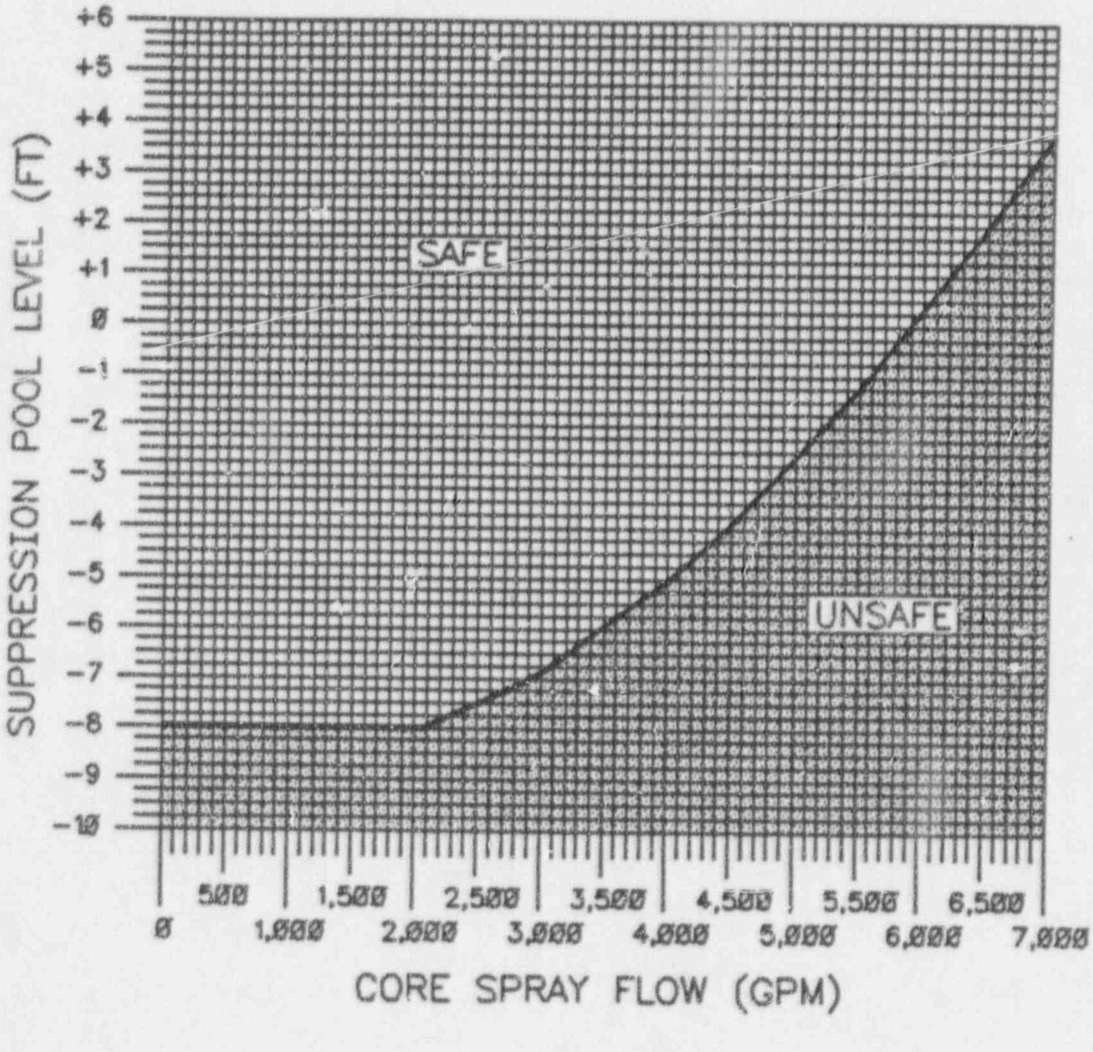
ATTACHMENT 5 (Cont'd)

FIGURE 8A
UNIT 2 SRV TAIL PIPE LEVEL LIMIT



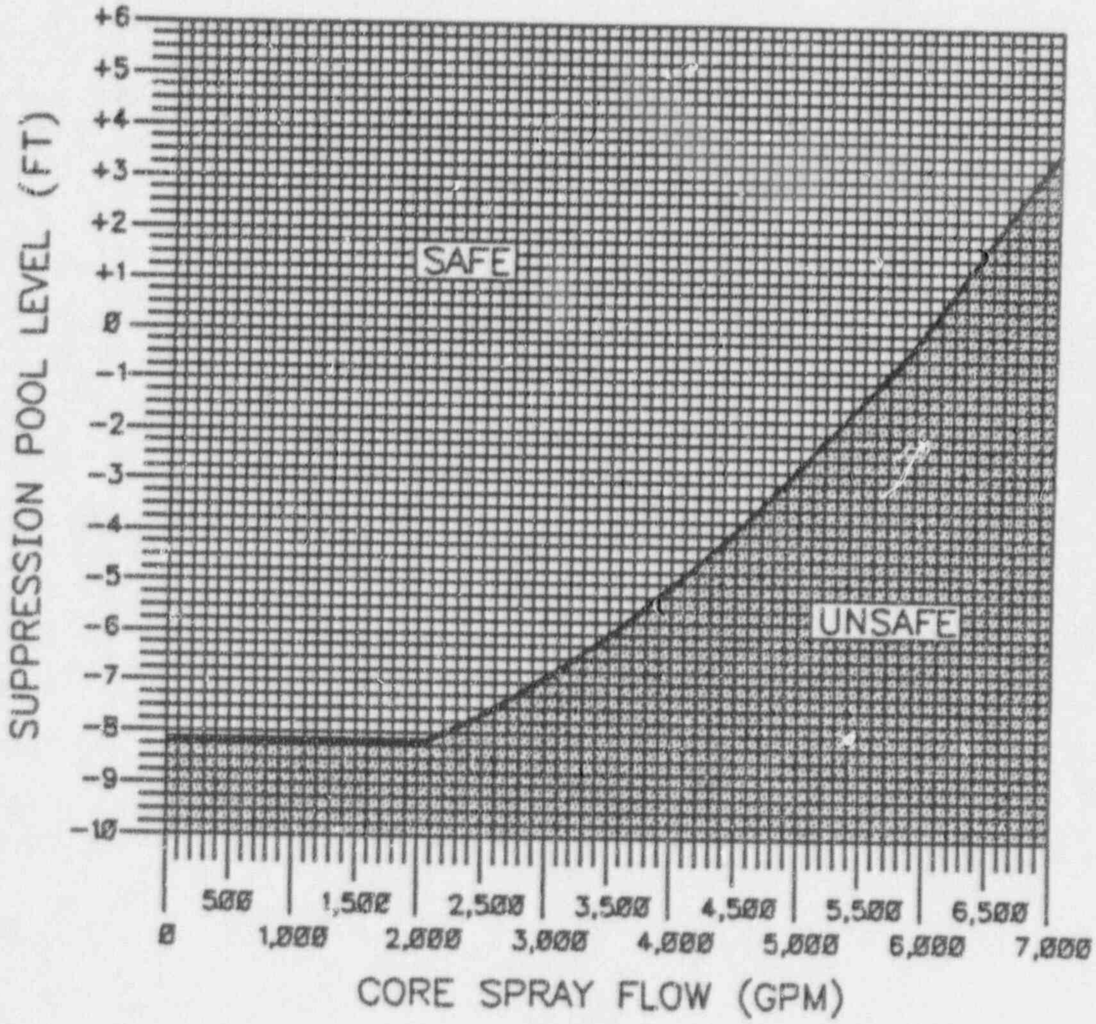
ATTACHMENT 5 (Cont'd)

FIGURE 9
UNIT 1 CORE SPRAY VORTEX LIMIT



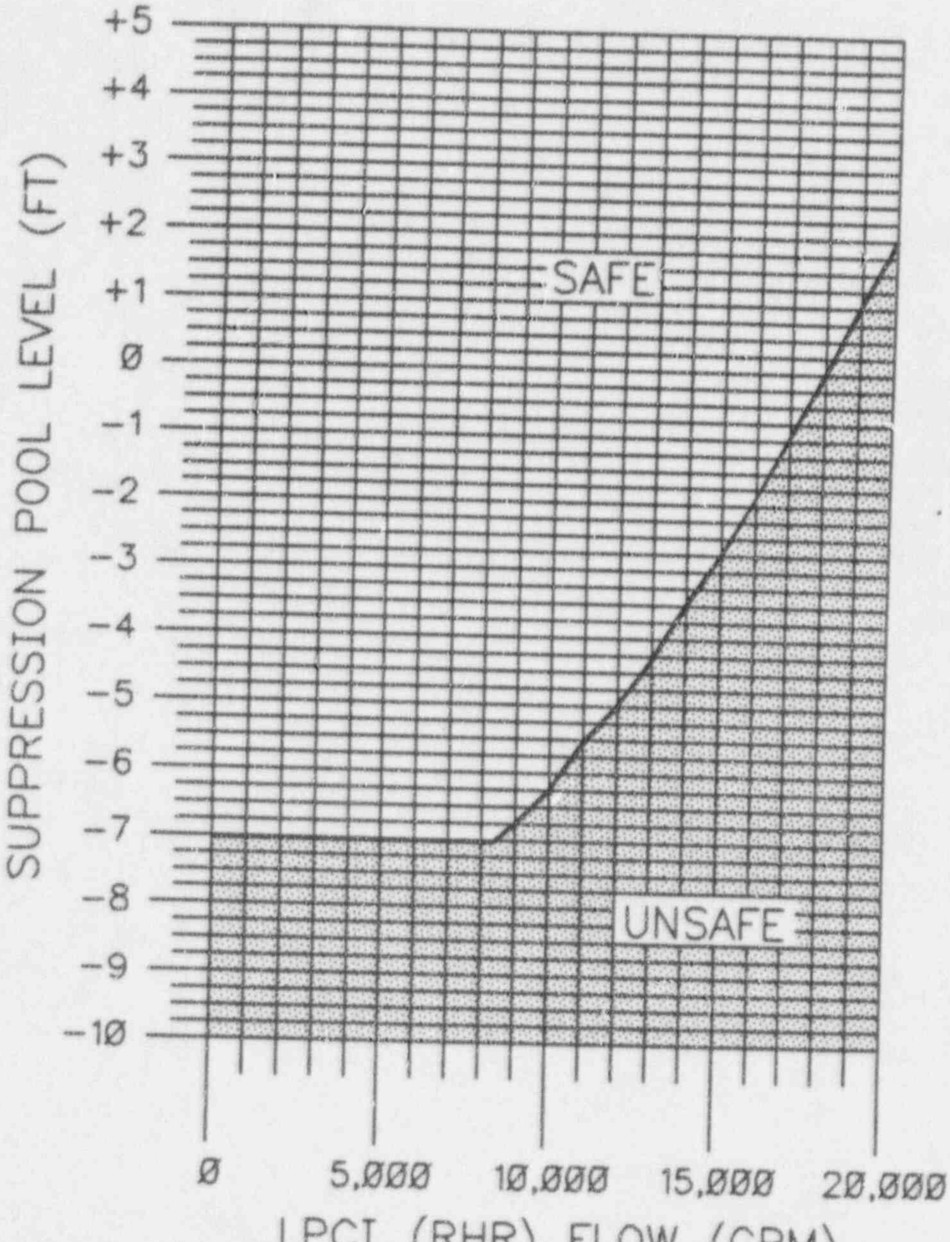
ATTACHMENT 5 (Cont'd)

FIGURE 10
UNIT 2 CORE SPRAY VORTEX LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 11
UNIT 1 RHR VORTEX LIMIT



ATTACHMENT 5 (Cont'd)

FIGURE 12
UNIT 2 RHR VORTEX LIMIT

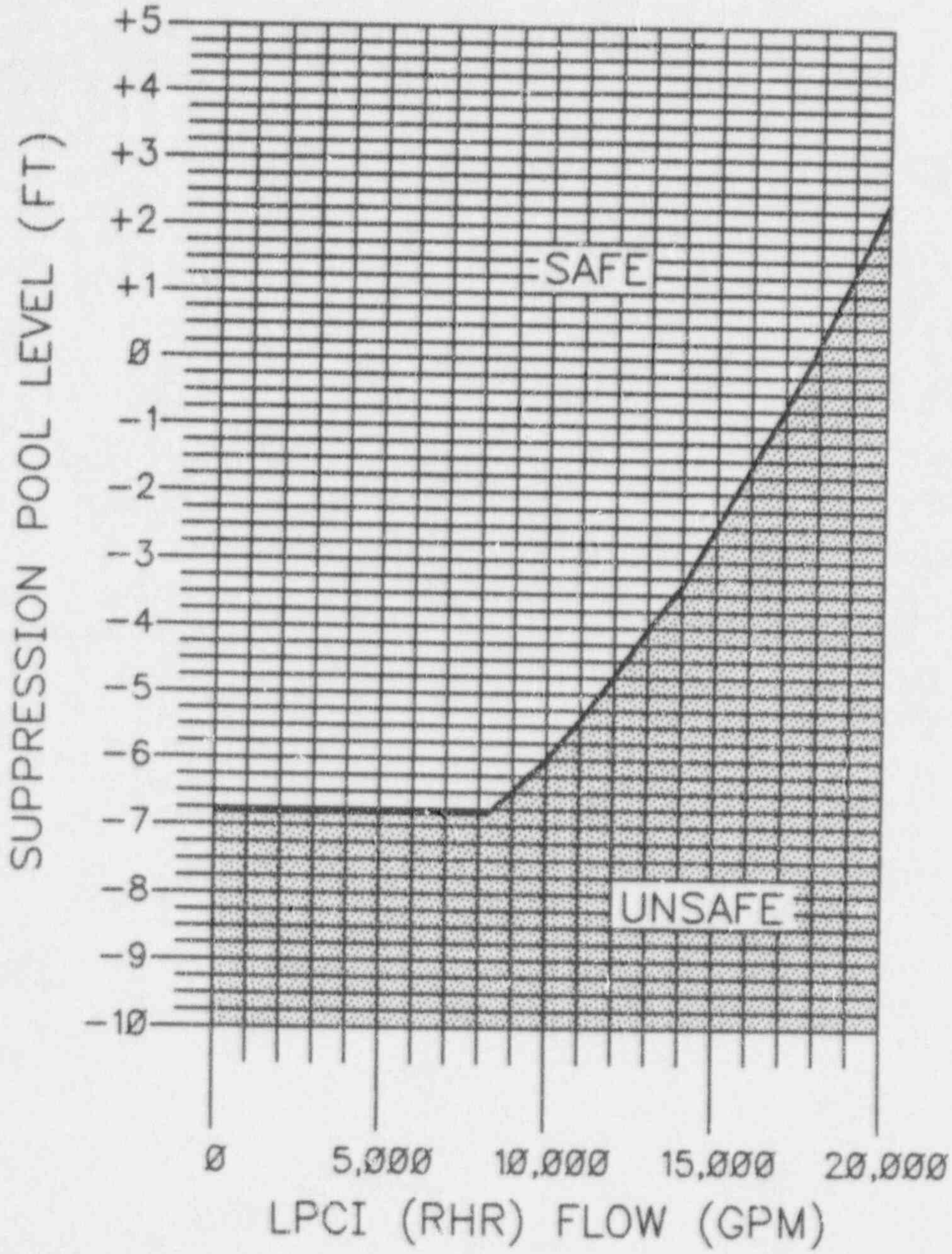


FIGURE 1

Thermal Power Limitations Map

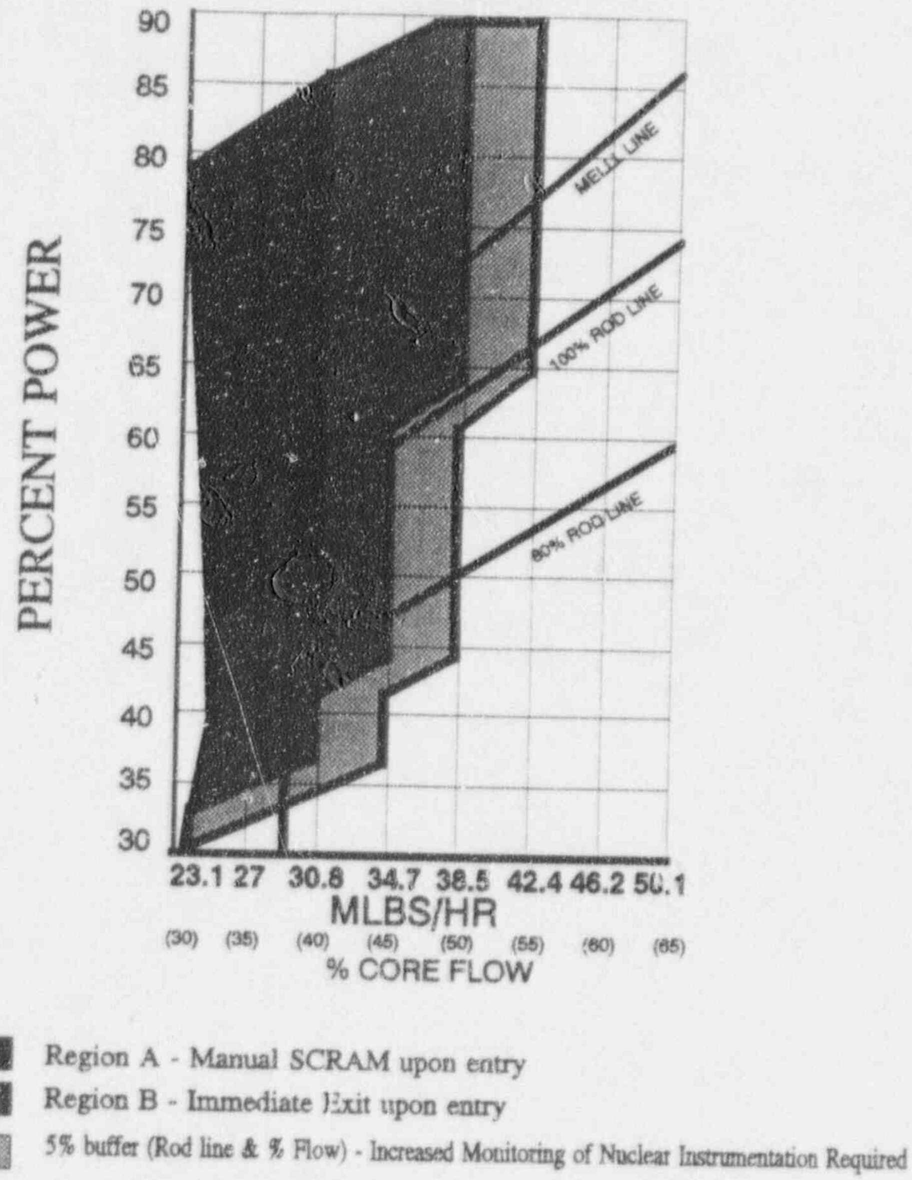


FIGURE 2

Rated Power vs. Core ΔP

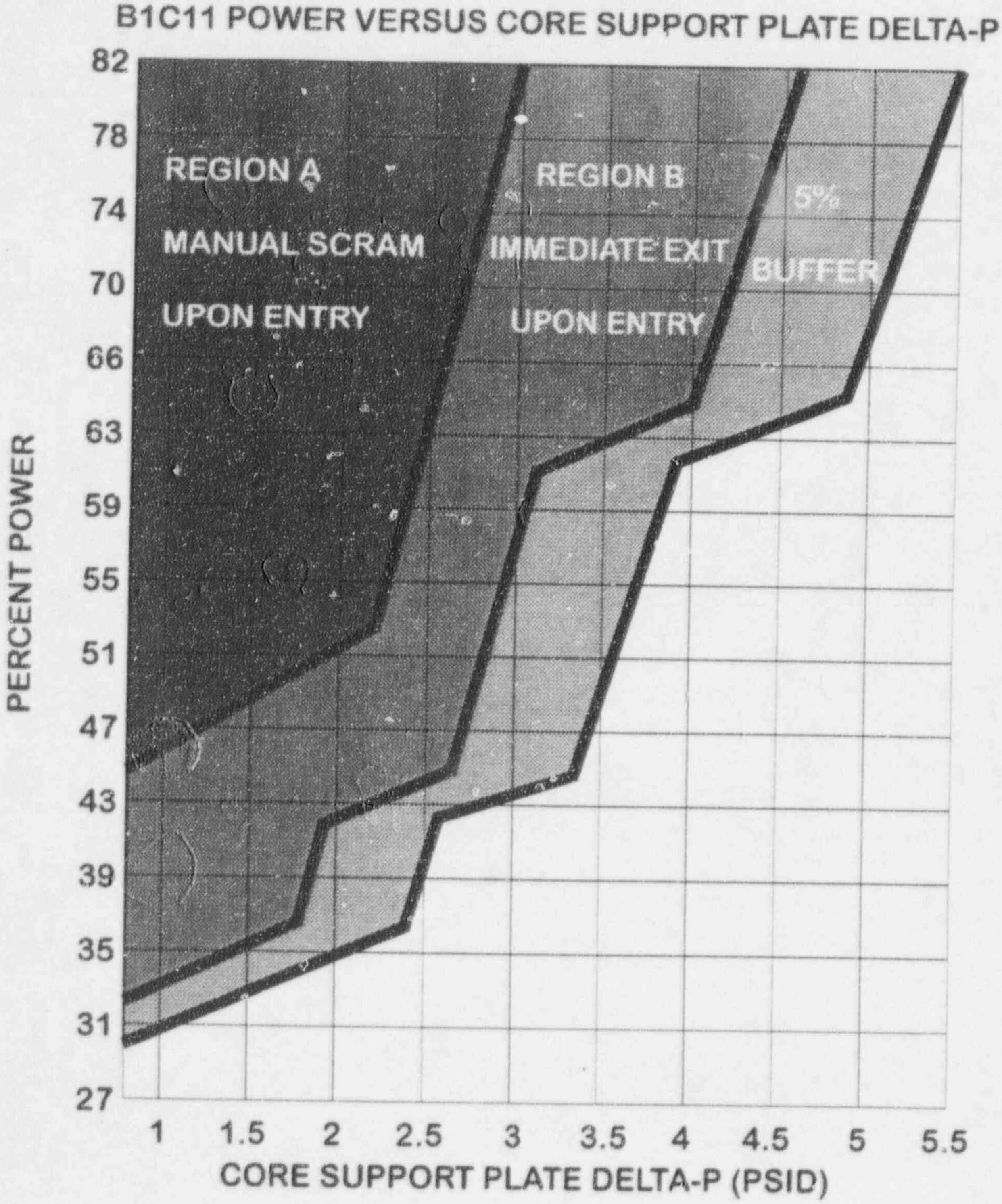
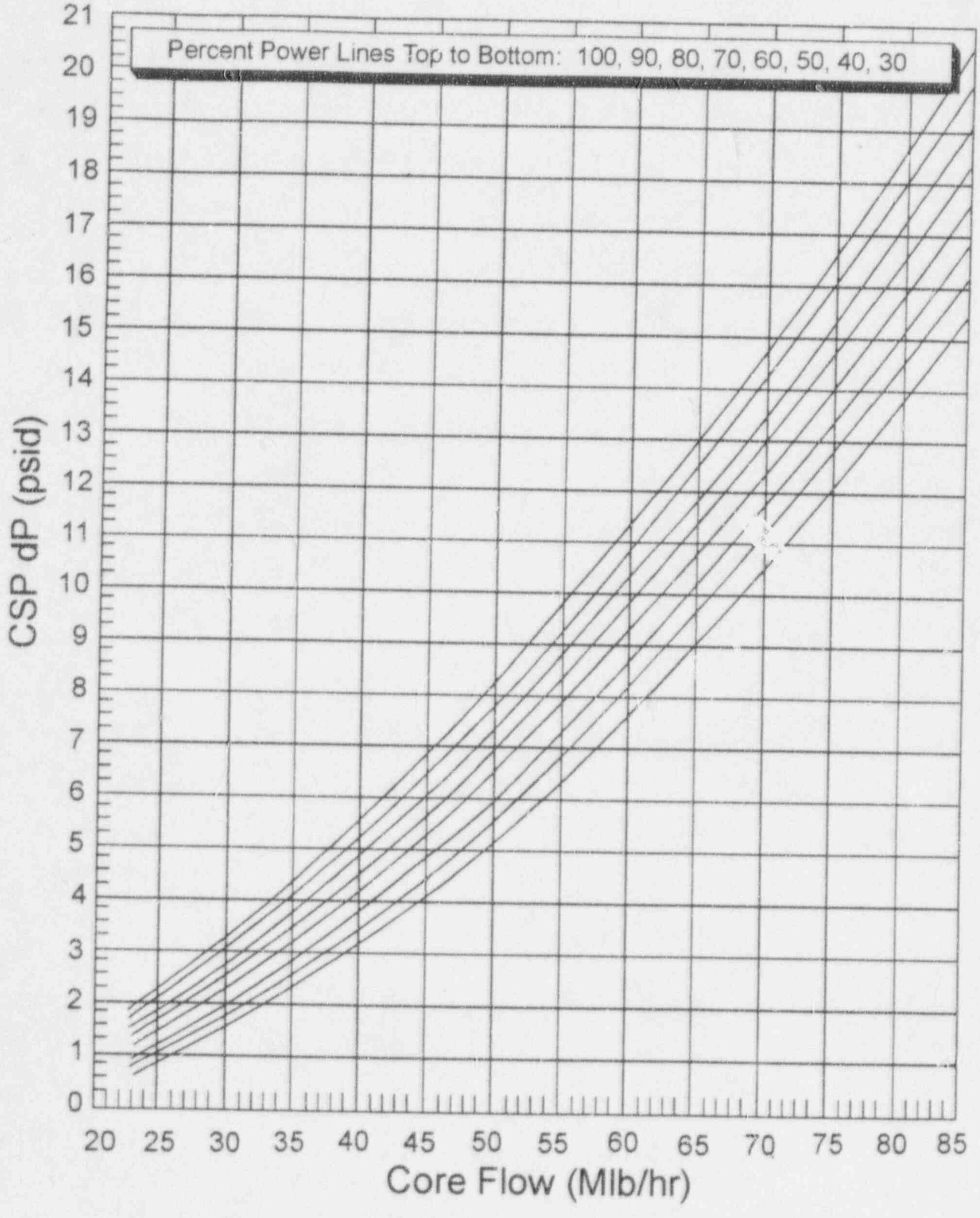


FIGURE 3

Estimated Total Core Flow vs. Core Support Plate Delta-P for B1C11



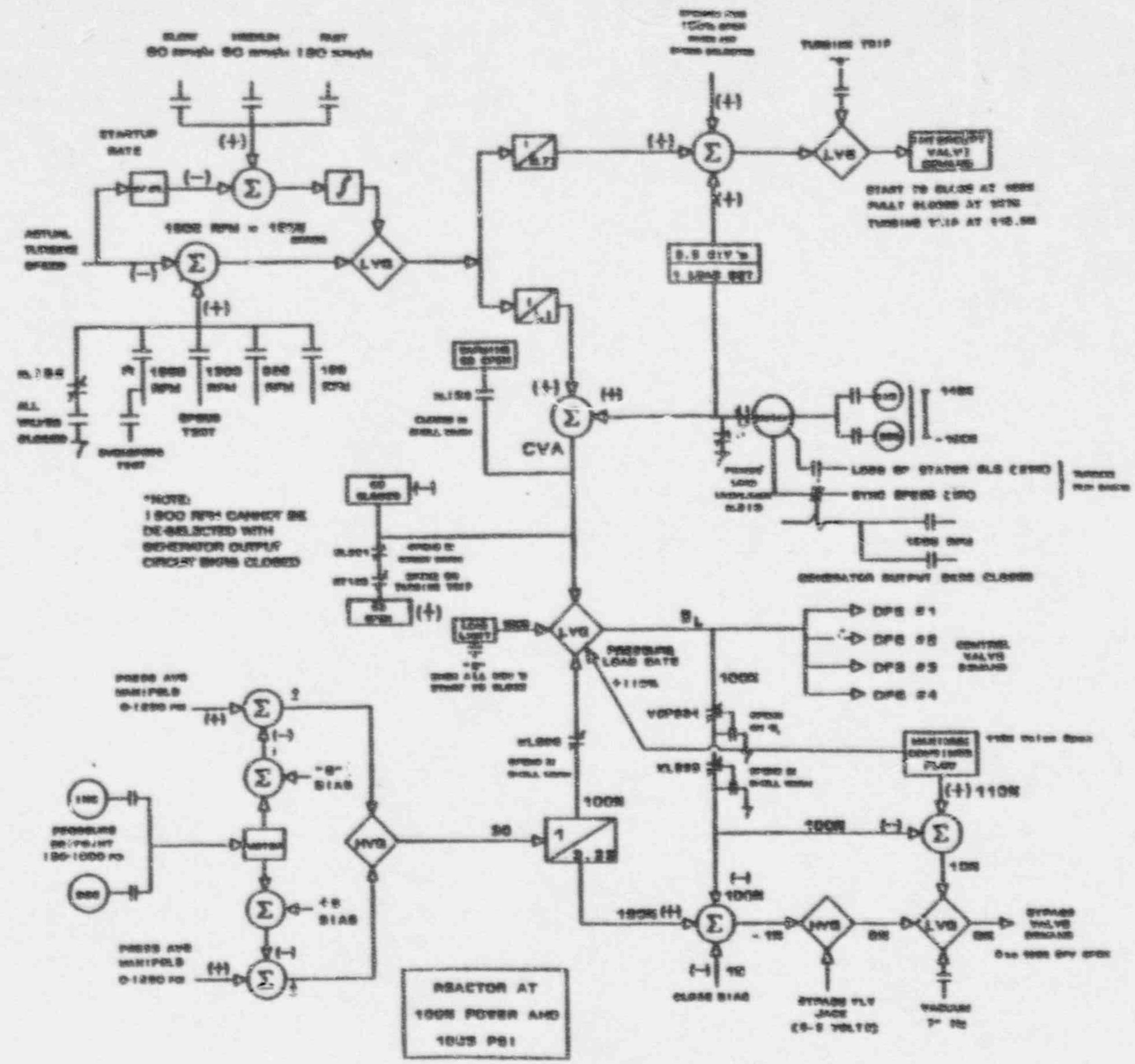


FIGURE 1
EHC ELECTRICAL BLOCK DIAGRAM

CP&L CAROLINA POWER & LIGHT COMPANY
BRUNSWICK NUCLEAR PLANT

PLANT OPERATING MANUAL

VOLUME XXI

ABNORMAL OPERATING PROCEDURE

UNIT
0

SELECT DISTRIBUTION

0AOP-15.0

LOSS OF SHUTDOWN COOLING

REVISION 8

EFFECTIVE DATE
6·27·96

Sponsor ____________ 6-20-96
Date

Approval ____________ 6-22-96
Manager - Operations CG Pardee x2723
Date

REVISION SUMMARY

Added caution in various locations describing the potential for a false low RPV water level signal under specific plant conditions which incorporate OPS AI 95-01883 Task 9. Deleted cautions, notes, and steps which referenced actions required for monitoring reactor coolant heatup/cooldown rates as this information is wholly covered in 1(2)PT-01.7 which is referenced in this procedure.

LIST OF EFFECTIVE PAGES

Page(s)	Revision
1-15	8

1.0 SYMPTOMS

1.1 *RHR SW PUMP 1A(2A) TRIP* (A-01 1-9) or *RHR SW PUMP 1C(2C) TRIP* (A-01 3-9) annunciator in alarm.

1.2 *RHR SW PUMP 1B(2B) TRIP* (A-03 1-8) or *RHR SW PUMP 1D(2D) TRIP* (A-03 3-8) annunciator in alarm.

1.3 *RHR PUMP 1A(2A) TRIP* (A-01 3-8) or *RHR PUMP 1C(2C) TRIP* (A-01 5-8) annunciator in alarm.

1.4 *RHR PUMP 1B(2B) TRIP* (A-03 3-7) or *RHR PUMP 1D(2D)* TRIP (A-03 5-7) annunciator in alarm.

1.5 *RHR HX A/B DISCH CLG WTR TEMP HI* (A-03 2-9) annunciator in alarm.

1.6 *RHR A/B DISCH & SUCT HDR PRESS HI* (A-03 3-9) annunciator in alarm.

1.7 *REACTOR VESS LO LEVEL TRIP* (A-05 2-6) annunciator in alarm.

1.8 Group 8 Isolation Valves close.

1.9 Increasing Reactor Coolant Temperature and/or Pressure.

R16 1.10 High NSW or CSW header pressure approaching pump shutoff head (approximately 90 psig).

R16 1.11 Unexplained changes in running RHRSW loop flow or pump discharge pressure.

2.0 AUTOMATIC ACTIONS

2.1 **IF** a Group 8 Isolation Signal exists (Low Level One or High Steam Dome Pressure), **THEN** the following will occur:

- *RHR SHUTDOWN COOLING OUTBOARD ISOLATION VALVE, E11-F008*, will close.

- *RHR SHUTDOWN COOLING INBOARD ISOLATION VALVE, E11-F009*, will close.

2.0 AUTOMATIC ACTIONS

- Loop A(B) *INBOARD INJECTION VALVE, E11-F015A(B)**, will close.

- The RHR Pump in service for Shutdown Cooling will trip on a loss of suction path.

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

None applicable

3.2 Supplementary Actions

CAUTION
R20 **IF** reactor coolant temperature is greater than 212°F, **AND** reactor water level has been raised to greater than 218 inches for 10 minutes or more, **THEN** a false RPV low level signal could result when the reference leg condensing pot N12A(B) nozzle is uncovered as level is subsequently lowered below 218 inches.

3.2.1 **IF** Shutdown Cooling has been lost due to a tripped RHR Pump, **THEN START** an RHR Pump in the loop being used for Shutdown Cooling.

3.2.2 **IF** forced circulation has been lost, **AND** natural circulation has **NOT** been established, **THEN RESTORE AND MAINTAIN** reactor vessel water level between 200" and 220" as read on *B21-LI-R605A(B)*, or as directed by Shift Superintendent based on plant conditions, until forced circulation is restored.

* Low Level One Only

3.2 Supplementary Actions

NOTE:	A Group 10 Isolation Signal will isolate the air supply to Reactor Head Vent Valves, *B21-F003* and *B21-F004*. These valves fail closed on a loss of air supply or power.

3.2.3 **DETERMINE** the position of Reactor Head Vent Valves, *B21-F003* and *B21-F004*.

1. **IF** vessel coolant temperature is greater than 212°F, or is indeterminate, **THEN ENSURE** Reactor Head Vent Valves, *B21-F003* and *B21-F004* are **CLOSED**.

CAUTION

Natural circulation cannot be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore cannot be utilized for vessel coolant temperature monitoring for indication of boiling. Under natural circulation conditions, reactor vessel pressure must be monitored for coolant temperature determination. If coolant temperature was initially less than 212°F, pressure must be closely monitored for indications of a trend of increasing pressure. If this trend is established, it must be assumed that 212°F has been exceeded, boiling is occurring, and a mode change has taken place.

3.2.4 **DIRECT** an operator to perform the following:

1. **MONITOR** reactor coolant heatup/cooldown in accordance with 1(2)PT-01.7 so that any unexpected trends can be promptly reported.

NOTE:	If the time to boiling in the reactor vessel **CANNOT** be determined, then it must be assumed that 212°F will be exceeded.

2. **OBTAIN** the approximate time to boiling in the reactor vessel based on current plant conditions (value should be in Daily Schedule Report).

3.2 Supplementary Actions

NOTE:	Secondary Containment Pressure Seal Work Permits are tracked in accordance with 0ENP-54.

3.2.5 **IF** it becomes apparent that Shutdown Cooling **CANNOT** be reestablished **OR** it has been determined that 212°F will be exceeded, **THEN:**

1. **DIRECT** Engineering to restore Secondary Containment prior to exceeding 212°F.
2. **RESTORE** Primary Containment prior to exceeding 212°F.
3. **CLOSE** Reactor Head Vent Valves, *B21-F003* and *B21-F004*.

3.2.6 **IF** the operating RHRSW loop has been lost, **THEN PERFORM** the following:

R16 1. **IF** unexplained changes in flow or pump discharge pressure are observed in the running RHRSW loop, **AND** NSW or CSW header pressure approaches pump shutoff head (approximately 90 psig), **THEN ENSURE** *UNIT 1(2) SERVICE WATER DISCHARGE OUTLET VALVE, 1(2)-SW-V442*, is open.

2. **IF** available, **THEN START** the idle RHRSW Booster Pump in the RHRSW loop being used for Shutdown Cooling.

3. **IF** the NSW or CSW Service Water Header has been lost, **THEN PLACE** the RHRSW loop in operation using the other Service Water header in accordance with 1(2)OP-43.

4. **IF NO** RHRSW Booster Pumps can be placed in operation, **THEN PLACE** RHRSW in operation with **NO** RHRSW Booster Pumps available in accordance with 1(2)OP-43.

5. **IF** Service Water is unavailable, **THEN PLACE** Fire Protection Water in operation to the Service Water Header in accordance with 0AOP-18.0.

3.2 Supplementary Actions

CAUTION

R20 **IF** reactor coolant temperature is greater than 212°F, **AND** reactor water level has been raised to greater than 218 inches for 10 minutes or more, **THEN** a false RPV low level signal could result when the reference leg condensing pot N12A(B) nozzle is uncovered as level is subsequently lowered below 218 inches.

3.2.7 **IF** the operating RHR loop in Shutdown Cooling has been lost, **AND** a Group 8 Isolation Signal does **NOT** exist, **AND NEITHER** RHR Pump in the RHR loop being used for Shutdown Cooling can be started, **THEN SHIFT** Shutdown Cooling loops in accordance with 1(2)OP-17.

3.2.8 **IF** the operating RHR loop in Shutdown Cooling has been lost, **AND** a Group 8 Isolation Signal exists, **THEN PERFORM** the following:

1. **RESTORE AND MAINTAIN** reactor water level in the previously established band in accordance with Shift Superintendent direction.

2. **REDUCE** reactor pressure below 125 psig in accordance with 0GP-05.

3. **ENSURE** RPS is energized.

4. **WHEN ALL** Group 8 Isolation signals have cleared, **THEN RESET** the Group 8 Isolation.

5. **PLACE** the RHR loop that was operating in Shutdown Cooling back in service in accordance with the following:

   a. **IF** piping cool down or drain down are a concern, **THEN GO TO** Step 3.2.8.6.

   b. **IF** piping cool down or drain down are **NOT** a concern, **THEN CONTINUE** with Step 3.2.8.5.c.

3.2 Supplementary Actions

c. **CLOSE** Loop A(B) *OUTBOARD INJECTION VALVE, E11-F017A(B).*

d. **OPEN** Loop A(B) *INBOARD INJECTION VALVE, E11-F015A(B).*

e. **OPEN** *RHR SHUTDOWN COOLING OUTBOARD ISOLATION VALVE, E11-F008.*

f. **OPEN** *RHR SHUTDOWN COOLING INBOARD ISOLATION VALVE, E11-F009.*

CAUTION
Failure to minimize RHR Pump operation while deadheaded may cause pump damage.

g. **START** an RHR Pump in the loop being used for Shutdown Cooling.

h. **SLOWLY THROTTLE OPEN** Loop A(B) *OUTBOARD INJECTION VALVE, E11-F017A(B)* to re-establish RHR loop conditions prior to the event.

i. **WHEN** RHR loop conditions have stabilized, **FULLY OPEN** Loop A(B) *OUTBOARD INJECTION VALVE, E11-F017A(B).*

j. **IF** the reactor coolant temperature is less than 212°F, **THEN ENSURE** Reactor Head Vent Valves, *B21-F003* and *B21-F004* are **OPEN**.

k. **MAINTAIN** RHR in Shutdown Cooling in accordance with 1(2)OP-17.

3.2 Supplementary Actions

NOTE:	Filling and venting may be required if the loop was idle for an extended period of time.

CAUTION

Piping may contain hot water at greater than 212°F due to previous Shutdown Cooling operation.

6. **IF** RHR has **NOT** been restored in Shutdown Cooling in accordance with the above steps, **THEN PLACE** the RHR loop that was operating in Shutdown Cooling back in service in accordance with 1(2)OP-17 as soon as conditions permit.

3.2.9 **IF** necessary to minimize reactor coolant temperature rise, **THEN PERFORM** one of the following feed and bleed combinations.

FEED	BLEED
COND/FW in accordance with 1(2)OP-32	RWCU Reject in accordance with 1(2)OP-14
CRD in accordance with 1(2)OP-08	Main Steam Line Drains in accordance with 1(2)OP-32
Core Spray in accordance with 1(2)OP-18	
LPCI in accordance with 1(2)OP-17	

3.2.10 **IF NEITHER** RHR loop can be placed in Shutdown Cooling, **THEN PLACE** the Condensate System in Condenser Cooling in accordance with 1(2)OP-32.

3.2.11 **IF ALL** of the above methods **CANNOT** maintain vessel coolant temperature below 212°F, **THEN INITIATE** alternate Shutdown Cooling with the SRVs as follows:

1. **ENSURE ALL** control rods are fully inserted.

3.2 Supplementary Actions

2. **IF** the Reactor Recirculation Pumps are running, **THEN PERFORM** the following:

    a. **RAISE AND MAINTAIN** reactor water level between 200" and 220" as read on *B21-LI-R605A(B)*, or as directed by Shift Superintendent based on plant conditions.

    b. **STOP** the running Reactor Recirculation Pumps in accordance with 1(2)OP-02.

3. **SHUT DOWN** the RHR loop that was operating in Shutdown Cooling in accordance with 1(2)OP-17.

4. **PLACE** one RHR loop in the Suppression Pool Cooling mode in accordance with 1(2)OP-17.

5. **IF** Suppression Pool temperature rises above 95°F, **THEN GO TO** 0EOP-02-PCCP, Primary Containment Control Procedure **AND PERFORM CONCURRENTLY** with this procedure.

6. **CLOSE** the following valves:

    a. Inboard and Outboard MSIVs (*B21-F022A-D* and *B21-F028A-D*).

    b. HPCI Inboard and Outboard Steam Supply Valves (*E41-F002* and *E41-F003*).

    c. RCIC Inboard and Outboard Steam Supply Valves (*E51-F007* and *E51-F008*).

    d. Inboard and Outboard Reactor Head Vent Valves (*B21-F003* and *B21-F004*).

    e. Inboard and Outboard Main Steam Line Drain Valves (*B21-F016* and *B21-F019*).

3.2 Supplementary Actions

7. **SELECT** one SRV based upon the desired cool down rate using the following table:

	RHR A/C	RHR B/D	CS A	CS B
HIGHEST COOLDOWN	B21-F013F B21-F013H	B21-F013A B21-F013B	B21-F013K	B21-F013E B21-F013L
	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D
	B21-F013A B21-F013B B21-F013K	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013A B21-F013B B21-F013K
	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J
LOWEST COOLDOWN	B21-F013E B21-F013L	B21-F013K	B21-F013A B21-F013B	B21-F013F B21-F013H

8. **PLACE** the control switch for the desired SRV to *OPEN*.

9. **RAISE AND MAINTAIN** reactor water level greater than 254 inches.

3.2 **Supplementary Actions**

NOTE:	The RHR pumps are preferred for injection.
NOTE:	Monitoring T_{SAT} per 1(2)PT-01.7 may **NOT** be valid under these special conditions due to reactor pressure **NOT** necessarily relating to T_{SAT}. Therefore, SRV tailpipe temperature recorder *B21-TR-R614* on Panel H12-P614, and/or ERFIS trending should be utilized for monitoring reactor coolant cool down rate.

10. **START** one RHR or Core Spray Pump.

11. **THROTTLE OPEN** the injection valve on the affected pump until the SRV opens.

12. **IF** reactor pressure **CANNOT** be maintained less than 164 psig above Suppression Chamber pressure, **THEN PLACE** another SRV control switch to *OPEN*.

13. **PERFORM** the following actions as necessary to maintain cool down rate less than 100°F per hour:

    a. **THROTTLE CLOSE** the injection valve on the affected pump until the desired SRV closes.

    b. **RECORD** reactor pressure at which the SRV closes.

    ________ psig

    c. **THROTTLE OPEN** the injection valve on the affected pump until the SRV reopens.

    d. **THROTTLE CLOSE** the injection valve on the affected pump until reactor pressure is 10 to 20 psig greater than the pressure at which the SRV closed in Step 3.2.11.13.b.

14. **IF** it is desired to **ADJUST** the cool down rate, **THEN CLOSE** the open SRV **AND OPEN** the next SRV that will **ADJUST** the cool down rate in the desired direction.

15. **REPEAT** Step 3.2.11.14 until vessel coolant and Suppression Pool temperature are within 100°F.

16. **CONTROL** Suppression Pool temperature as necessary to maintain vessel coolant temperature above 75°F.

3.2 Supplementary Actions

17. **WHEN** a normal method of Shutdown Cooling can be established, **THEN SHUT DOWN** alternate Shutdown Cooling as follows:

    a. **STOP** the ECCS pump(s) used for vessel injection.

    b. **WHEN** the SRV(s) that were opened have closed, **THEN PLACE** the control switch for the SRV(s) to *CLOSE* **OR** *AUTO*.

    c. **IF** the reactor coolant temperature is less than 212°F, **THEN OPEN** Reactor Head Vent Valves, *B21-F003* and *B21-F004*.

CAUTION

R20 **IF** reactor coolant temperature is greater than 212°F, **AND** reactor water level has been raised to greater than 218 inches for 10 minutes or more, **THEN** a false RPV low level signal could result when the reference leg condensing pot N12A(B) nozzle is uncovered as level is subsequently lowered below 218 inches.

    d. **RESTORE AND MAINTAIN** reactor water level between 200" and 220", or as directed by the Shift Superintendent, based on plant conditions.

    e. **WHEN** directed by the Shift Superintendent, **THEN SHUT DOWN** the RHR loop used for Suppression Pool Cooling in accordance with 1(2)OP-17.

4.0 GENERAL DISCUSSION

An extended loss of the decay heat removal function can lead to elevated vessel coolant temperatures, localized or bulk coolant boiling and potentially result in a depletion of reactor coolant and eventual uncovering of the core. If no forced circulation exists during a Loss of Shutdown Cooling event, natural circulation must be established. Natural circulation however, cannot be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore cannot be utilized for vessel coolant temperature monitoring for heatup rate determination or indication of boiling.

In addition, if RWCU is not in service with suction from the bottom head, vessel bottom head drain temperature cannot be used for verification of Tech Spec 3.4.6.1 (Pressure/Temperature Curves) compliance for cooldowns.

4.0 GENERAL DISCUSSION

Under natural circulation conditions, reactor vessel pressure must be monitored for vessel coolant temperature determination. If vessel coolant temperature is less than 212°F, pressure must be closely monitored for indications of a trend of increasing pressure. If this trend is established, it must be assumed that 212°F has been exceeded, boiling is occurring, and a mode change has taken place.

R20 If reactor coolant temperature is greater than 212°F, and reactor water level has been raised to greater than 218 inches for 10 minutes or more, then a false RPV low level signal could result when the reference leg condensing pot N12A(B) nozzle is uncovered as level is subsequently lowered below 218 inches. This false signal is the result of water exiting the nozzle and condensing pots at the same time steam is re-entering the reference leg. This counter flow condition sets up the conditions conducive to steam bubble creation and collapse. This causes a momentary upward pressure spike in the reference leg, which gives a momentary indicated false signal to the transmitters involved.

While irradiated fuel remains in the reactor vessel during an outage, maintaining the decay heat removal function remains a key to shutdown safety. The risk associated with a loss of decay heat removal event is dependent on a number of factors, including the decay heat load present and the existing plant configuration. Outage risk assessment will ensure that adequate contingency plans are in place prior to reducing decay heat removal capability.

This procedure addresses a loss of normal decay heat removal capability during shutdown conditions. The procedure provides contingencies for the following methods of decay heat removal:

- RHRSW Loop Failure
- RHR Loop Failure
- Condenser Cooling Failure
- Feed and Bleed Combinations
- Alternate Shutdown Cooling with SRVs

This procedure also provides contingencies for restoring primary and secondary containment, and initial emergency actions for a loss of Shutdown Cooling.

Industry events have occurred which demonstrate that use of the SRVs to steam to the Suppression Pool is a viable method of decay heat removal. Makeup requirements can be supplied by a single CRD Pump in this mode of cooling. Outage risk assessment should prevent the need for ever using this mode of cooling.

5.0 REFERENCES

5.1 Regulatory Guide 1.33, Quality Assurance Program Requirements (Operations) (November 1972), Appendix A, Item F.8

5.2 ANSI Standard N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, Section 5.3.9.2, Item (3)

5.3 Technical Specification 3.4.6.1, Pressure/Temperature Limits

5.4 Technical Specification 3.6.1.1, Primary Containment Integrity

5.5 Technical Specification 3.6.5.1, Secondary Containment Integrity

5.6 FSAR Section 5.4.7, 7.4.3, 15.2.7

5.7 NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management

5.8 0AOP-18.0, Nuclear Service Water System Failure

5.9 1(2)OP-02, Reactor Recirculation System Operating Procedure

5.10 1(2)OP-08, Control Rod Drive Hydraulic System Operating Procedure

5.11 1(2)OP-14, RWCU System Operating Procedure

5.12 1(2)OP-17, RHR System Operating Procedure

5.13 1(2)OP-18, Core Spray System Operating Procedure

5.14 1(2)OP-32, Condensate and Feedwater System Operating Procedure

5.15 1(2)OP-43, Service Water System Operating Procedure

R16 5.16 IER #92-21-03 (IFI); FACTS #93B9034

5.17 0EOP-02-PCCP, Primary Containment Control Procedure

5.18 0ENP-54, Building Ventilation Pressure Control Program

5.19 1(2)PT-01.7, Heatup/Cooldown Monitoring

R20 5.20 CR 95-01883, False RPV Water Level Low Level 1 Signals

Section 1 - Primary Containment Water Level Calculation

1. Suppression Pool water level instruments may be used for levels up to +2 feet.

   a. CAC-LI-2601-1 (XU-51)

   b. CAC-LR-2602 (XU-51)

2. IF Suppression Chamber pressure is less than 75 psig, THEN CALCULATE Primary Containment water level as follows:

   P_1 = Suppression Chamber pressure plus head of water

   CAC-PI-1257-2A (XU-51)
   CAC-PI-1257-2B (P601)

   P_2 = Drywell pressure at greater than 85 ft elevation

   CAC-PI-1230 (P601)

$$PC_{wl} = \frac{2.3 \text{ feet}}{\text{psi}} (P_1 - P_2) + 5.3 \text{ feet}$$

TIME					
P_1(psig)					
P_2(psig)					
$P_1 - P_2$					
x 2.3	x 2.3	x 2.3	x 2.3	x 2.3	x 2.3
+ 5.3	+ 5.3	+ 5.3	+ 5.3	+ 5.3	+5.3
PC_{wl}(ft)					

Section 1 (Continued)

3. IF Suppression Chamber pressure is greater than 75 psig, THEN CALCULATE Primary Containment water level as follows:

P_1 = Primary Containment pressure plus head of water

CAC-PI-4176 (XU-51)
CAC-PR-1257-1 (XU-51)

P_2 = Drywell pressure at greater than 85 ft elevation

CAC-PI-1230 (P601)

P_1 measured using CAC-PI-4176

$$PC_{wl} = \frac{2.3 \text{ feet}}{\text{psi}} (P_1 - P_2) + 28.5 \text{ feet}$$

P_1 measured using CAC-PR-1257-1

$$PC_{wl} = \frac{2.3 \text{ feet}}{\text{psi}} (P_1 - P_2) + 30.5 \text{ feet}$$

Using PI-4176

TIME					
P_1(psig)					
P_2(psig)					
P_1 - P_2					
x 2.3	x 2.3	x 2.3	x 2.3	x 2.3	x 2.3
+ 28.5	+ 28.5	+ 28.5	+ 28.5	+ 28.5	+ 28.5
PC_{wl}					

Section 1 (Continued)

Using PR-1257-1

TIME					
P_1(psig)					
P_2(psig)					
P_1 - P_2					
x 2.3	x 2.3	x 2.3	x 2.5	x 2.3	x 2.3
+ 30.5	+ 30.5	+ 30.5	+ 30.5	+ 30.5	+ 30.5
PC_{wl}(ft)					

4. IF Drywell water level is greater than 60.5 feet, THEN USE CAC DRYWELL WATER LEVEL INDIC, CAC-LI-1216 on P601.

CP&L

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK NUCLEAR PLANT

C
Continuous
Use

DATE COMPLETED ____________
UNIT________ % PWR_____ GMWE__________
SUPERVISOR____________
REASON FOR TEST (check one or more):
_____ Routine surveillance
_____ OWP #____________
_____ WR/JO #____________
_____ Other (explain)____________

FREQUENCY:

Each reactor coolant system heatup/cooldow-.

PLANT OPERATING MANUAL

VOLUME X

PERIODIC TEST

UNIT
2

BNP RECIPIENT ID
-068-
CONTROLLE!

2PT-01.7

HEATUP/COOLDOWN MONITORING

REVISION 0

RECEIVED BY BNP
MAY 13 1996
NUCLEAR DOCUMENT CONTROL

EFFECTIVE DATE
5/14/96

Sponsor ____________ 5/9/96
Date

Approval ____________ 5-10-96
Manager - Operations
Date

REVISION SUMMARY

This procedure was issued to provide a method to determine compliance with the requirements of Technical Specifications 3.4.6.1, Items a and b, except during inservice hydrostatic or leak testing. This surveillance satisfies Technical Specification 4.4.6.1.1 Surveillance Requirement during Reactor Coolant System heatup and cooldown.

LIST OF EFFECTIVE PAGES

Page(s)	Revision
1-11	0

1.0 PURPOSE

1.1 This surveillance is performed to determine compliance with the requirements of Technical Specifications 3.4.6.1, Items a and b, except during inservice hydrostatic or leak testing. This surveillance satisfies Technical Specification 4.4.6.1.1 Surveillance Requirement during Reactor Coolant System heatup and cooldown.

1.2 This test involves data collection of certain plant parameters, and confirmation that reactor coolant system pressure and temperature are maintained within limits.

2.0 REFERENCES

2.1 Technical Specifications

2.2 GE SIL No. 430, Reactor Pressure Vessel Temperature Monitoring

2.3 GE SIL No. 251, Control of RPV Bottom Head Temperatures

2.4 GE SIL No. 251 Supplement 1, BWR Vessel Bottom Head Coolant Temperature

2.5 0GP-02, Approach to Criticality and Pressurization of the Reactor

2.6 0GP-05, Unit Shutdown

2.7 2OP-17, Residual Heat Removal System Operating Procedure

2.8 0AOP-15.0, Loss of Shutdown Cooling

2.9 0EOP-01-RSP, Reactor Scram Procedure

3.0 PRECAUTIONS AND LIMITATIONS

3.1 **IF** PPC Display 860, RPV HEATUP/COOLDOWN MONITOR, is available, **THEN** the value of **DT/HR** to be recorded on Attachment 2 is the as-displayed numerical value.

3.2 **IF** T_{SAT} is above 212°F, **OR** in natural circulation, and PPC Display 860, RPV HEATUP/COOLDOWN MONITOR is **NOT** available, **THEN** coolant temperature is to be determined from the Steam Tables utilizing reactor steam dome pressure as read on RTGB instruments.

3.0 PRECAUTIONS AND LIMITATIONS

3.3 **WHEN** T_{SAT} is less than 212°F, **OR** when RPV pressure monitoring instrumentation is **NOT** valid, **THEN** coolant temperature is to be determined by one of the following methods:

3.3.1 Reactor Recirc Pump running **AND** loop is **NOT** isolated from the Reactor, **THEN** use Recirculation Suction Temperatures read on B32-TR-R650.

3.3.2 RHR Pump is running in Shutdown Cooling mode with the Heat Exchanger aligned as follows:

3.3.2.1 RHR HX in service: Use RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614.

3.3.2.2 RHR HX **NOT** in service: Use RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614.

3.4 Bottom Head temperature during heatup and cooldown may be determined in a number of ways depending on the status of the RWCU System and the Reactor Recirc Pumps.

3.4.1 During heatup, C12-TR-R018, Channel 151, Bottom Head metal temperature on Panel H12-P007, is the preferred source due to the metal temperature response lagging the coolant temperature response. **IF** Channel 151 is unavailable, **THEN** alternate methods can be used depending on RWCU status as follows:

- RWCU in service: Use Vessel Bottom Drain Temperature as read on G31-TI-R607 on Panel P603, **OR** C12-TR-R018 Channel 153, on Panel H12-P007.

- RWCU **NOT** in service **AND** Reactor Recirc Pumps 2A **AND** 2B running at ≥ 20% speed: Use Recirc Suction Temperature as read on B32-TR-R650.

3.0 PRECAUTIONS AND LIMITATIONS

3.4.2 During cooldown, Bottom Head coolant temperature is the preferred source due to coolant temperature response leading vessel metal temperature response.

- **IF** RWCU is in service, **THEN** Vessel Bottom Drain coolant temperature can be determined from G31-TI-R607, **OR** C12-TR-R018, Channel 153.
- **IF** RWCU is **NOT** in service, **THEN** Recirc suction temperature as read on B32-TR-R650 may be utilized, **BUT ONLY IF** both Recirc Pumps are running at greater than or equal to 20% speed.
- **IF** RWCU **AND** Recirc are **NOT** in service, **THEN** use C12-TR-R018, Channel 151.

4.0 PREREQUISITES

None

5.0 SPECIAL TOOLS AND EQUIPMENT

Steam Tables

6.0 ACCEPTANCE CRITERIA

6.1 **WHEN** the following criteria are met, **THEN** this procedure may be considered satisfactory:

6.1.1 Calculated heatup or cooldown in any one-hour period does **NOT** exceed 100°F. This is the value of Differential Temperature/Hour (**DT/HR**) recorded on Attachment 2 or 3.

6.1.2 Operation is to the right and/or below limiting lines of Technical Specification Figures 3.4.6.1-1 or 3.4.6.1-2, as applicable. This is determined by verification that the **BOTTOM HEAD** value recorded on Attachment 2 or 3 is to the right and/or below limiting lines for vessel pressure of the appropriate Technical Specification Figures (Attachment 4 or 5).

7.0 PROCEDURAL STEPS

NOTE:	Data should be recorded at an increased frequency of at least once per 10-15 minutes during transient conditions such as scram, loss of heat sink, uncontrolled depressurization, etc.

7.1 **COMPLETE** the applicable portions of the Data Sheet of Attachment 2 or 3, at least once per 30 minutes. ☐

7.2 **INITIAL** on the appropriate Data Sheet to indicate that both acceptance criteria are met. ☐

7.3 **IF** either acceptance criterion is **NOT** met, **OR** limits are being approached, **THEN IMMEDIATELY NOTIFY** the Unit SCO. ☐

7.4 **CONTINUE** monitoring until determined unnecessary by the Unit SCO. ☐

Certification and Review Form

2PT-01.7

General Comments and Recommendations ______________________________

____

	Initials	Name (Print)
Test procedure performed by:	________	______________________
	________	______________________
	________	______________________
	________	______________________

Exceptions to satisfactory performance ______________________________

____

Corrective action required ______________________________

____

Test procedure <u>has</u> been satisfactorily completed

Unit SCO ______________________ ____________
Signature Date

Test procedure <u>has not</u> been satisfactorily completed

Unit SCO ______________________ ____________
Signature Date

Test has been reviewed by

Shift Superintendent ______________________ ____________
Signature Date

Data Sheet (T_{SAT} > 212°F or Natural Circulation)

DATE/TIME	T_{SAT} °F CURRENT	T_{SAT} °F MAX/MIN in hour	DT/HR* +/- 100°F / Initials (Criteria met)	BOTTOM HEAD** °F / Initials (Criteria met)
			/	/
			/	/
			/	/
			/	/
			/	/
			/	/
			/	/
			/	/
			/	/
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			/	/
			/	/
			/	/

This Attachment may be duplicated as needed for periodic monitoring.

* IF PPC DISPLAY 860, RPV HEATUP/COOLDOWN MONITOR is available, **THEN RECORD** the displayed value of **DT/HR** and NA the T_{SAT} columns. **IF** PPC DISPLAY 860 is not available, **THEN DT/HR** is calculated as the larger of **T_{SAT} CURRENT** minus the maximum or minimum T_{SAT} in the previous hour. The value may be positive or negative dependent on plant heatup or cooldown.

** **BOTTOM HEAD** temperature during a plant cooldown is determined from G31-TI-R607, Pt. 5 or C12-TR-R018, Ch. 153, **IF** RWCU is in service. **BOTTOM HEAD** temperature during a plant heatup is determined from C12-TR-R018, Ch. 151 (preferred), or either C12-TR-R018, Ch. 153, or G31-TI-R607, Pt. 5, **IF** RWCU is in service. **IF** RWCU is **NOT** in service, **BOTTOM HEAD** temperature monitoring should be performed using C12-TR-R018, Ch. 151, **OR** Recirc suction temperature **IF** both Recirc Pumps are running at greater than or equal to 20% speed.

Data Sheet (T_{SAT} < 212°F or Invalid Pressure Instruments)

DATE/TIME	RECIRC A °F	RECIRC B °F	RHR SDC °F	DT/HR* +/- 100°F / Initials (Criteria met)	BOTTOM HEAD** °F / Initials (Criteria met)
				/	/
				/	/
				/	/
				/	/
				/	/
				/	/
				/	/
				/	/
				/	/
				/	/
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				/	/
				/	/
				/	/
				/	/
				/	/
				/	/
				/	/

This Attachment may be duplicated as needed for periodic monitoring.

* **DT/HR** is calculated as the largest change in either valid **RECIRC A** or **B** temperatures in the previous hour, or if in **SDC**, the largest change in the appropriate RHR HX temperature in the previous hour. The value may be positive or negative dependent on plant heatup or cooldown.

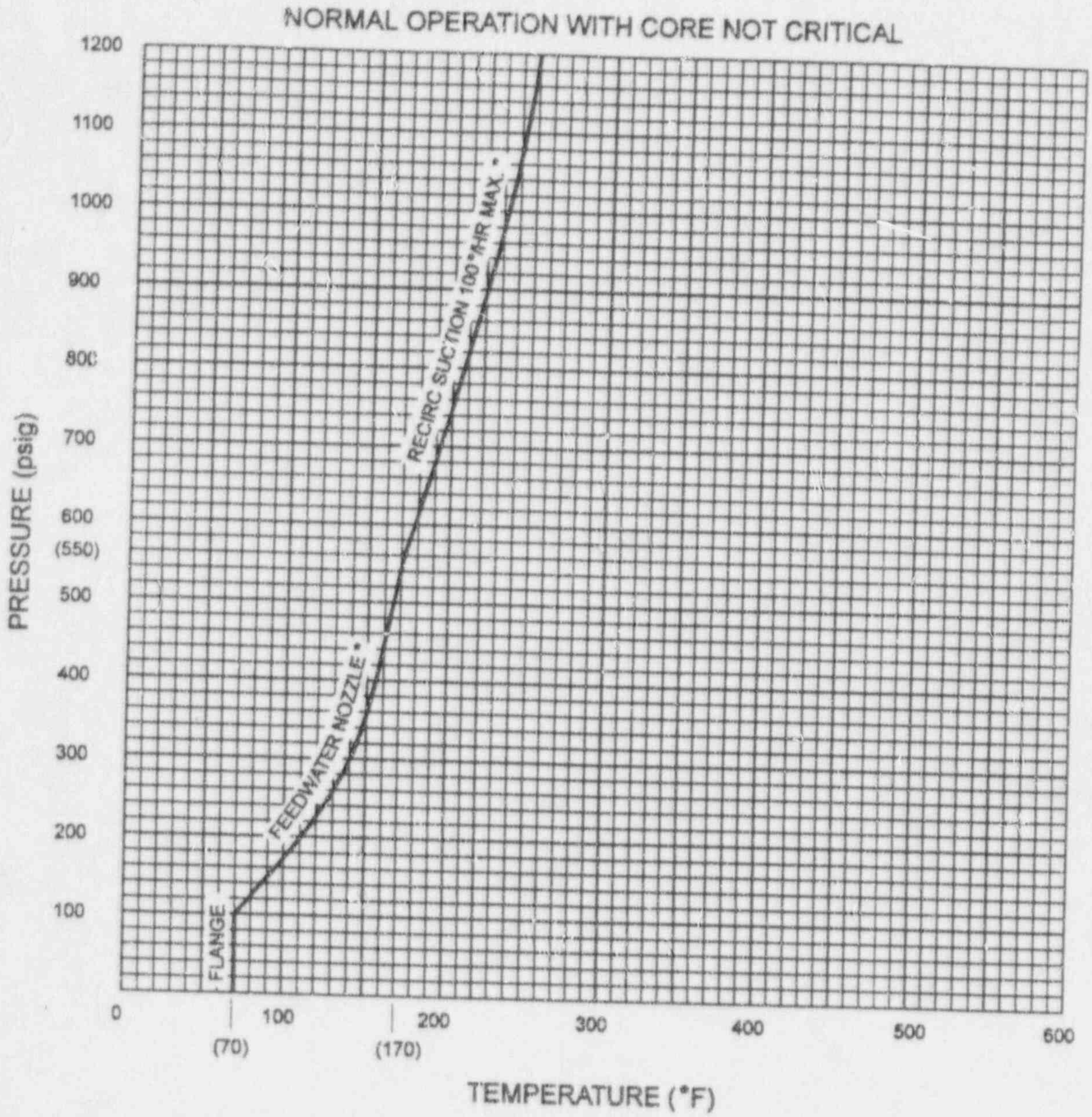
** **BOTTOM HEAD** temperature during a plant cooldown is determined from G31-TI-R607, Pt. 5 or C12-TR-R018, Ch. 153, **IF** RWCU is in service. **BOTTOM HEAD** temperature during a plant heatup is determined from C12-TR-R018, Ch. 151 (preferred), or either C12-TR-R018, Ch. 153, or G31-TI-R607, Pt. 5, **IF** RWCU is in service. **IF** RWCU is **NOT** in service, **BOTTOM HEAD** temperature monitoring should be performed using C12-TR-R018, Ch. 151, **OR** Recirc suction temperature **IF** both Recirc Pumps are running at greater than or equal to 20% speed.

ATTACHMENT 4

Page 1 of 1

Tech Spec Figure 3.4.6.1-1 Amendment 172

Pressure-Temperature Limits Reactor Vessel

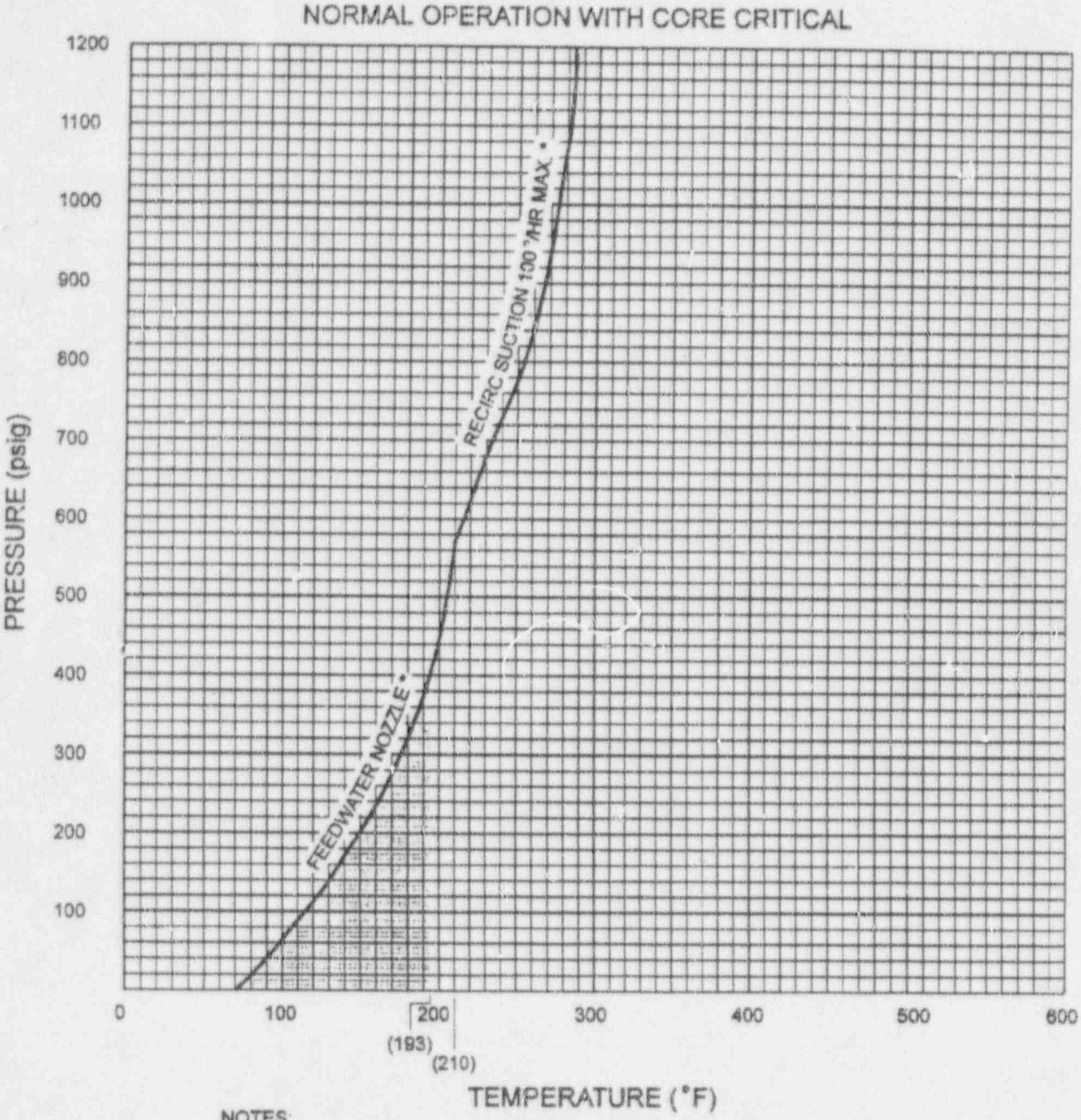


NOTES:

1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES

ATTACHMENT 5

Tech Spec Figure 3.4.6.1-2 Amendment 172
Pressure-Temperature Limits Reactor Vessel

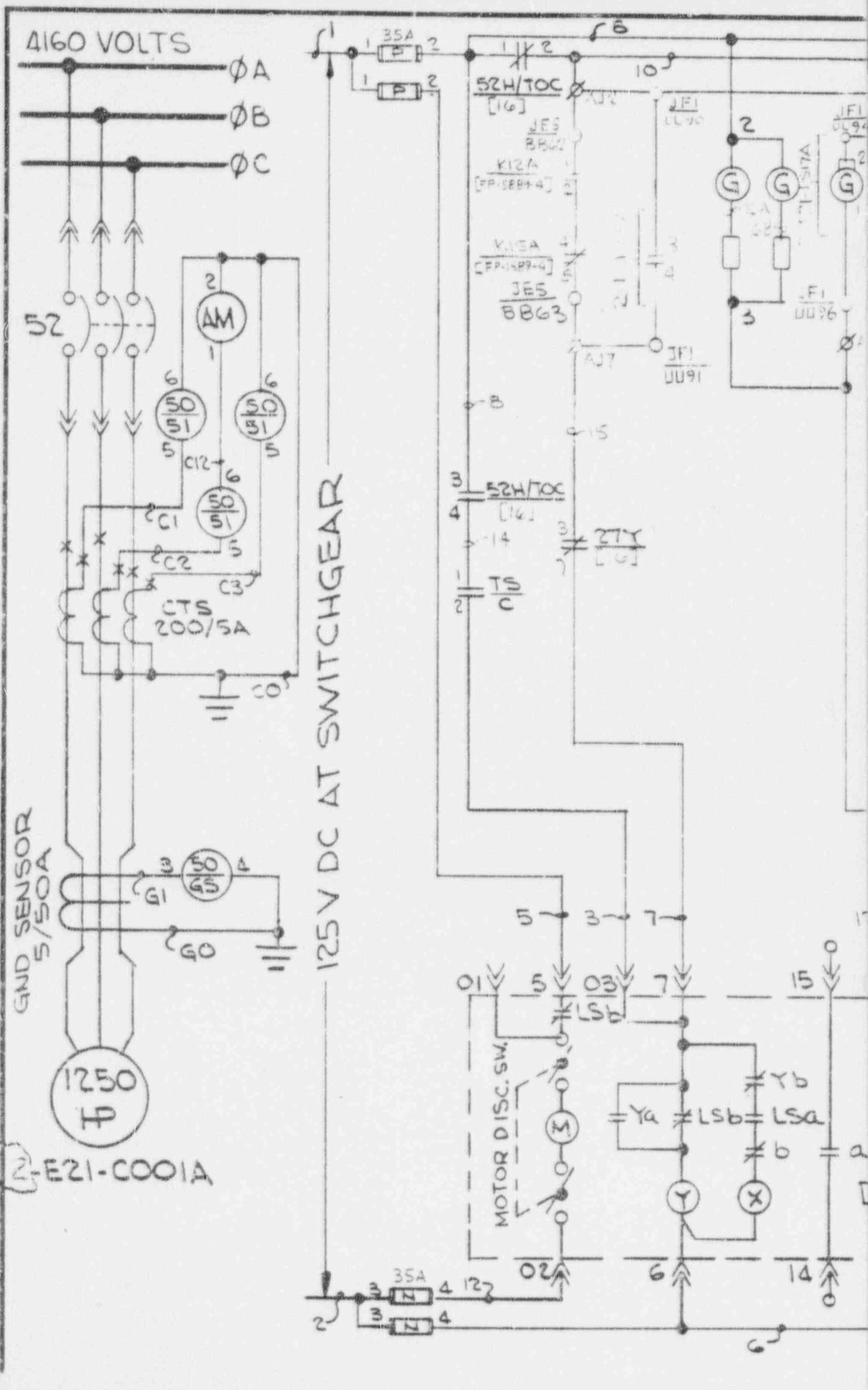


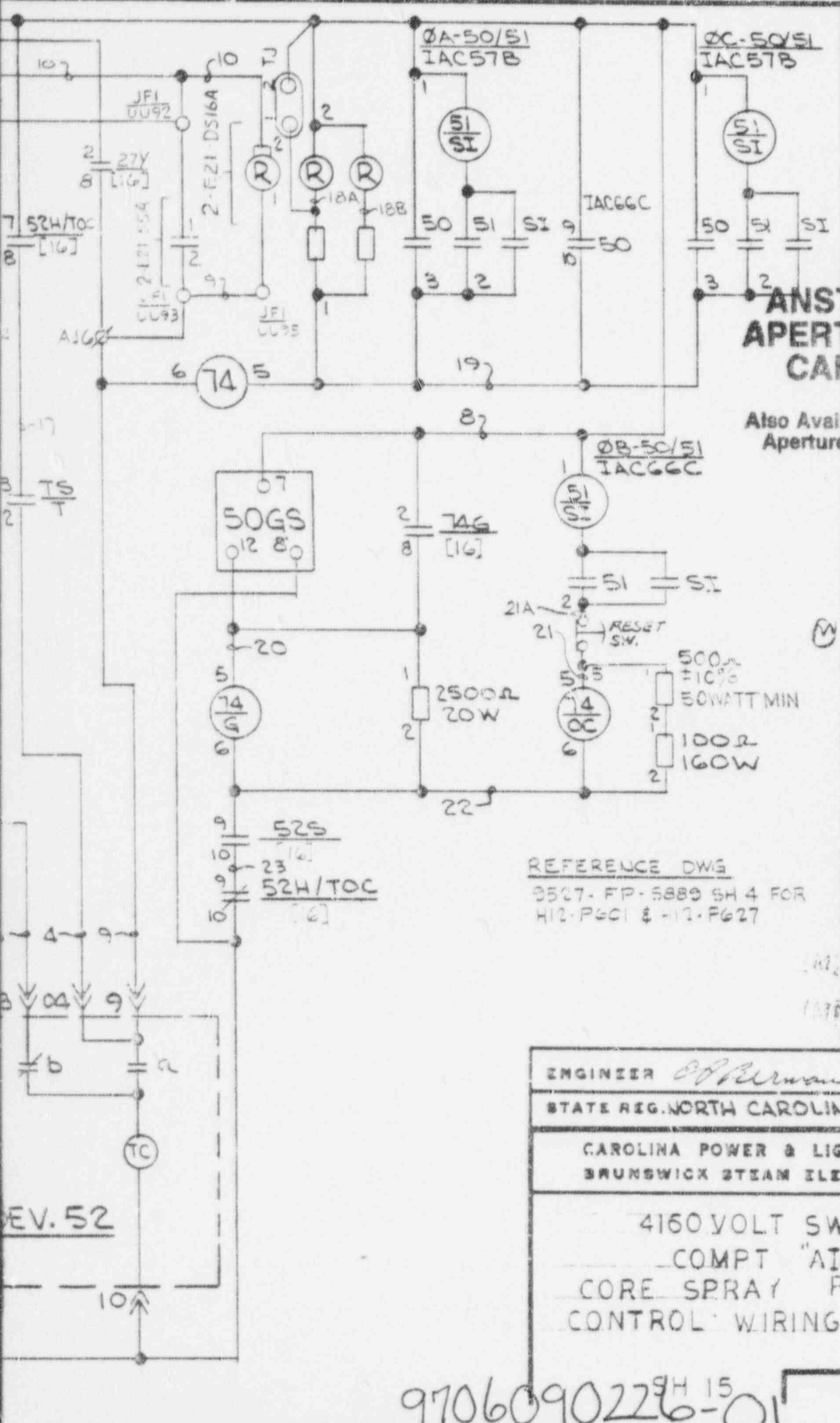
NOTES:
1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
4. OPERATION IN SHADED AREA PERMITTED ONLY WHEN WATER LEVEL IS WHITHIN NORMAL RANGE FOR POWER OPERATION.

DRAWING TRANSFERRED TO
THE CUSTODY OF CP&L
REV 5 UC 33762

6

2-E21-C001A





REV	DESCRIPTION
6	12-8-88 REVISED PER PDC-88-0064 WB CTJ AJS BF
5	2-12-79 AS BUILT PER PM 77-138 B.P.
4	AS BUILT PER F.E.C.P. 1127 2-10-76 AFB
3	REV'D PER UBX-7601 3-11-75 RLK
2	9-19-74 J.M. DELETED TERM, AJ3 & WIRE No 11
1	REVISED JUMPER 6-7-73 JE
0	FIRST ISSUE

ENGINEER [signature] DATE 10-13-72

STATE REG. NORTH CAROLINA No. 6037

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

4160 VOLT SWGR E3
COMPT "AI6"
CORE SPRAY PUMP 2A
CONTROL WIRING DIAGRAM

SH 15

LL-09113

9706090226-01

REFERENCES

FOR WRITTEN TEST

SRO ONLY

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery or until the reactor is placed in an OPERATIONAL CONDITION in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified applicability state shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL CONDITIONS required to comply with ACTION requirements.

3.0.5 When a system, subsystem, train, component, or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystems(s), train(s), component(s), and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, and in at least COLD SHUTDOWN within the following 30 hours. This specification is not applicable in Conditions 4 or 5.

RETYPED TECH. SPECS.
Updated Thru. Amend.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable state shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

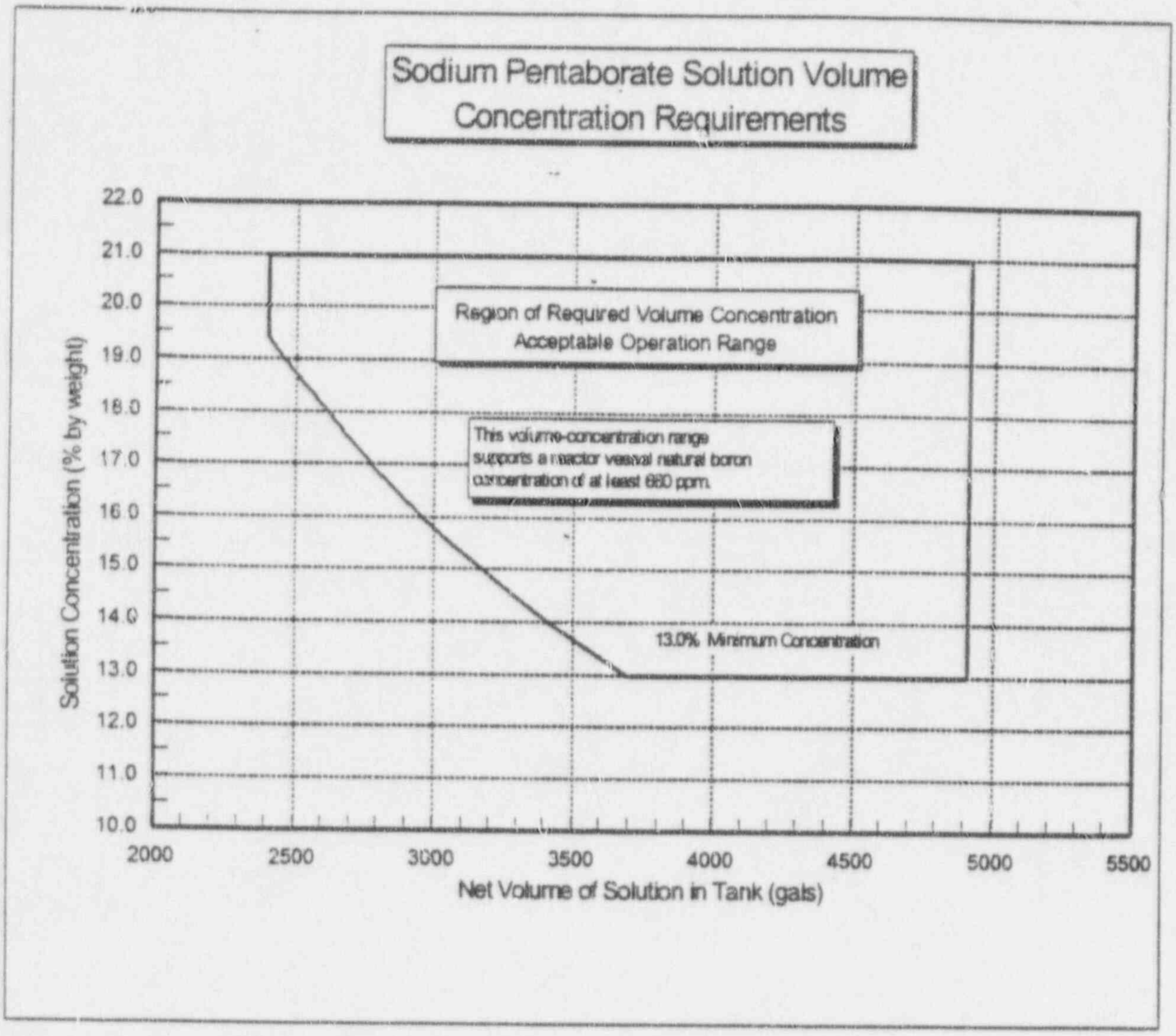
c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

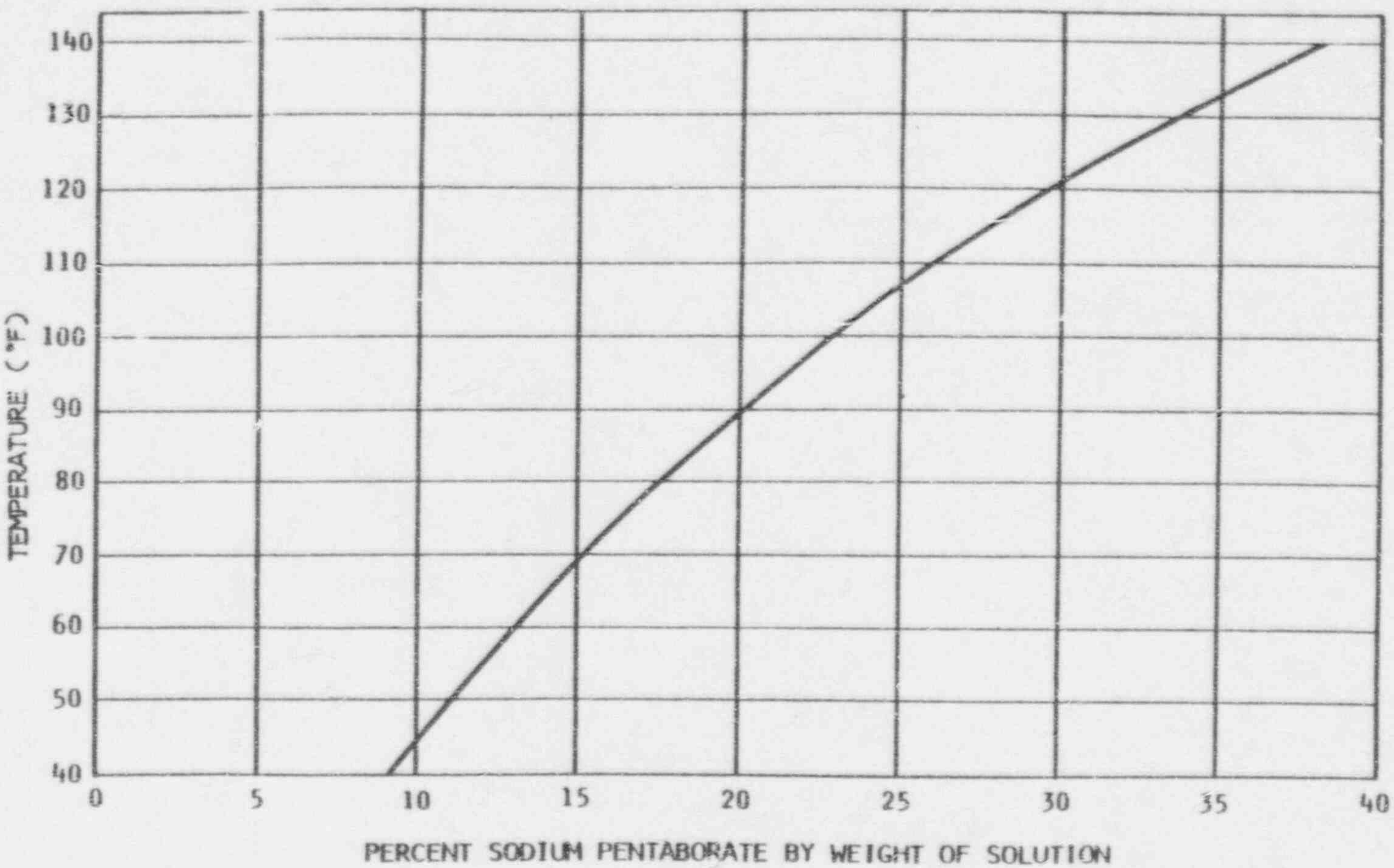
d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods & personnel & sample expansion inlcuded in this letter.

SODIUM PENTABORATE SOLUTION VOLUME CONCENTRATION REQUIREMENTS
FIGURE 3.1.5-1





SATURATION TEMPERATURE OF SODIUM
PENTABORATE SOLUTION

FIGURE 3.1.5-2

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. For any isolation actuation Trip Function with less than the Minimum Number of OPERABLE Channels per Trip System required by Table 3.3.2-1:

   1. Within one hour, verify sufficient channels remain OPERABLE or are placed in the tripped condition* to maintain automatic isolation actuation capability for the Trip Function, and

   2. Place the inoperable channel(s) in the tripped condition* within:

      a) 12 hours for trip functions common to RPS Instrumentation, and

      b) 24 hours for trip functions not common to RPS Instrumentation

   Otherwise, take the ACTION required by Table 3.3.2-1.

c. Deleted.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

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* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.3.2-1 for the Trip Function shall be taken.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation function# shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation function.

Radiation monitors are exempt from response time testing.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - 1. Low, Level 1	2, 6	2	1, 2, 3	20
	8	2	1, 2, 3	27
2. Low, Level 3	1	2	1, 2, 3	20
b. Drywell Pressure - High	2, 6	2	1, 2, 3	20
c. Main Steam Line 1. (Deleted)				
2. Pressure - Low	1[j]	2	1	22
3. Flow - High	1[j]	2/line	1	22
4. Flow - High	1[j]	2	2, 3	21
d. Main Steam Line Tunnel Temperature - High	1[j]	2[d]	1, 2, 3	21
e. Condenser Vacuum - Low	1	2	1, 2[e]	21
f. Turbine Building Area Temperature - High	1[j]	4[d]	1, 2, 3	21
g. Main Stack Radiation - High	(h)	1	1, 2, 3	28
h. Reactor Building Exhaust Radiation - High	6	1	1, 2, 3	20

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Exhaust Radiation - High	(1)	1	1, 2, 3, 5, and *	23
	6	1	1, 2, 3	20
b. Drywell Pressure - High	(1)	2	1, 2, 3	23
	2, 6	2	1, 2, 3	20
c. Reactor Vessel Water Level - Low, Level 2	(1)	2	1, 2, 3	23
	3	2	1, 2, 3	24
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	3	1	1, 2, 3	24
b. Area Temperature - High	3	2	1, 2, 3	24
c. Area Ventilation Δ Temperature - High	3	2	1, 2, 3	24
d. SLCS Initiation	3 (f)	NA	1, 2	24
e. Reactor Vessel Water Level - Low, Level 2	3	2	1, 2, 3	24
f. Δ Flow - High - Time Delay	NA	1	1, 2, 3	24
g. Piping Outside RWCU Rooms Area Temperature - High	3	1	1, 2, 3	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. CORE STANDBY COOLING SYSTEMS ISOLATION				
a. High Pressure Coolant Injection System Isolation				
1. HPCI Steam Line Flow - High	4	1	1, 2, 3	25
2. HPCI Steam Line Flow - High Time Delay Relay	NA	1	1, 2, 3	25
3. HPCI Steam Supply Pressure - Low	4 7(k)	2 1	1, 2, 3 1, 2, 3	25 25
4. HPCI Steam Line Tunnel Temperature - High	4	2	1, 2, 3	25
5. Bus Power Monitor	NA(g)	1/bus	1, 2, 3	26
6. HPCI Turbine Exhaust Diaphragm Pressure - High	4	2	1, 2, 3	25
7. HPCI Steam Line Ambient Temperature - High	4	1	1, 2, 3	25
8. HPCI Steam Line Area Δ Temperature - High	4	1	1, 2, 3	25
9. HPCI Equipment Area Temperature - High	4	1	1, 2, 3	25
10. Drywell Pressure - High	7(k)	1	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued)				
b. Reactor Core Isolation Cooling System Isolation				
1. RCIC Steam Line Flow - High	5	1	1, 2, 3	25
2. RCIC Steam Line Flow - High Time Delay Relay	NA	1	1, 2, 3	25
3. RCIC Steam Supply Pressure - Low	5	2	1, 2, 3	25
	$9^{(k)}$	1	1, 2, 3	25
4. RCIC Steam Line Tunnel Temperature - High	5	2	1, 2, 3	25
5. Bus Power Monitor	$NA^{(g)}$	1/bus	1, 2, 3	26
6. RCIC Turbine Exhaust Diaphragm Pressure - High	5	2	1, 2, 3	25
7. RCIC Steam Line Ambient Temperature - High	5	1	1, 2, 3	25
8. RCIC Steam Line Area Δ Temperature - High	5	1	1, 2, 3	25
9. RCIC Equipment Room Ambient Temperature - High	5	1	1, 2, 3	25
10. RCIC Equipment Room Δ Temperature - High	5	1	1, 2, 3	25
11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay	NA	1	1, 2, 3	25
12. Drywell Pressure - High	$9^{(k)}$	1	1, 2, 3	25

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM NUMBER OPERABLE CHANNELS PER TRIP SYSTEM(b)(c)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. SHUTDOWN COOLING SYSTEM ISOLATION				
a. Reactor Vessel Water Level - Low, Level 1	2, 6	2	1, 2, 3	20
	8	2	1, 2, 3	27
b. Reactor Steam Dome Pressure - High	8(i)	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTIONS

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 21 - Be in at least STARTUP with the main steam line isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - In OPERATIONAL CONDITIONS 1, 2, or 3, establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

In OPERATIONAL CONDITION 5 or when handling irradiated fuel in the secondary containment:

1) Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour;

2) Otherwise, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS, or activities that could reduce the SHUTDOWN MARGIN.

ACTION 24 - Isolate the reactor water cleanup system.

ACTION 25 - Close the affected system isolation valves and declare the affected system inoperable.

ACTION 26 - Verify power availability to the bus at least once per 12 hours.

ACTION 27 - Deactivate the shutdown cooling supply and reactor vessel head spray isolation valves in the closed position until the reactor steam dome pressure is within the specified limits.

ACTION 28 - Close the affected isolation valves within 14 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

NOTES

* When handling irradiated fuel in the secondary containment.

(a) See Specification 3.6.3.1, Table 3.6.3-1 for valves in each valve group.

(b) When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONs may be delayed as follows:

(1) For up to 2 hours for Trip Functions with a design that provides only one channel per trip system.

(2) For up to 6 hours for all Trip Functions, provided the Trip Function maintains isolation actuation capability.

(c) Deleted.

(d) A channel is OPERABLE if 2 of 4 instruments in the channel are OPERABLE.

(e) With reactor steam pressure ≥ 500 psig.

(f) Closes only RWCU outlet isolation valve.

(g) Alarm only.

(h) Isolates containment purge and vent valves.

(i) Does not isolate E11-F015A,B.

(j) Does not isolate B32-F019 or B32-F020.

(k) Valve isolation depends upon low steam supply pressure coincident with high drywell pressure.

(l) Secondary containment isolation dampers as listed in Table 3.6.5.2-1.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The Emergency Core Cooling System (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific ECCS function.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low, Level 3	4	1, 2, 3, 4, 5	30
b. Reactor Steam Dome Pressure - Low (Injection Permissive)	4	1, 2, 3, 4, 5	30
c. Drywell Pressure - High	4	1, 2, 3	30
d. Time Delay Relay	1/pump	1, 2, 3, 4, 5	31
e. Bus Power Monitor[d]	1/bus	1, 2, 3, 4, 5	32
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Drywell Pressure - High	4	1, 2, 3	30
b. Reactor Vessel Water Level - Low, Level 3	4	1, 2, 3, 4[b], 5[b]	30
c. Reactor Vessel Shroud Level (Drywell Spray Permissive)	1/valve	1, 2, 3, 4[b], 5[b]	31
d. Reactor Steam Dome Pressure - Low (Injection Permissive)			
1. RHR Pump Start and LPCI Injection Valve Actuation	4	1, 2, 3, 4[b], 5[b]	30
2. Recirculation Loop Pump Discharge Valve Actuation	4	1, 2, 3, 4[b], 5[b]	30
e. RHR Pump Start - Time Delay Relay	1/pump	1, 2, 3, 4[b], 5[b]	31
f. Bus Power Monitor[d]	1/bus	1, 2, 3, 4[b], 5[b]	32

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
3. HIGH PRESSURE COOLANT INJECTION SYSTEM			
a. Reactor Vessel Water Level - Low, Level 2	4	1, 2, 3	30
b. Drywell Pressure - High	4	1, 2, 3	30
c. Condensate Storage Tank Level - Low	2[c]	1, 2, 3	33
d. Suppression Chamber Water Level - High	2[c]	1, 2, 3	33
e. Bus Power Monitor[d]	1/bus	1, 2, 3	32
4. AUTOMATIC DEPRESSURIZATION SYSTEM			
a. ADS Inhibit Switch	2	1, 2, 3	36
b. Reactor Vessel Water Level - Low, Level 3	4	1, 2, 3	36
c. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3	36
d. ADS Timer	2	1, 2, 3	36
e. Core Spray Pump Discharge Pressure - High (Permissive)	4	1, 2, 3	36
f. RHR (LPCI MODE) Pump Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	36
g. Bus Power Monitor[d]	1/bus	1, 2, 3	32

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	1/bus	1/bus	1/bus	1, 2, 3, 4(e), 5(e)	34
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	3/bus	2/bus	2/bus	1, 2, 3, 4(e), 5(e)	35

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTIONS

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. Within 1 hour, verify sufficient channels remain OPERABLE or are placed in the tripped condition to maintain automatic ECCS actuation capability for the Trip Function, and

b. Within 24 hours, place all inoperable channels that do not cause the Trip Function to occur in the tripped condition.

Otherwise, declare the associated ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, verify bus power availability at least once per 12 hours or declare the associated ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour or declare the HPCI system inoperable.

ACTION 34 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 35 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, verify within one hour that a sufficient number of channels remain OPERABLE to maintain actuation capability of either ADS Trip System A or ADS Trip System B and restore the inoperable channels to OPERABLE status within 24 hours. Otherwise, declare ADS inoperable.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

NOTES

(a) When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONs may be delayed for up to 6 hours provided the Trip Function or the redundant Trip Function maintains ECCS actuation capability.

(b) Not applicable when two core spray system subsystems are OPERABLE per Specification 3.5.3.1.

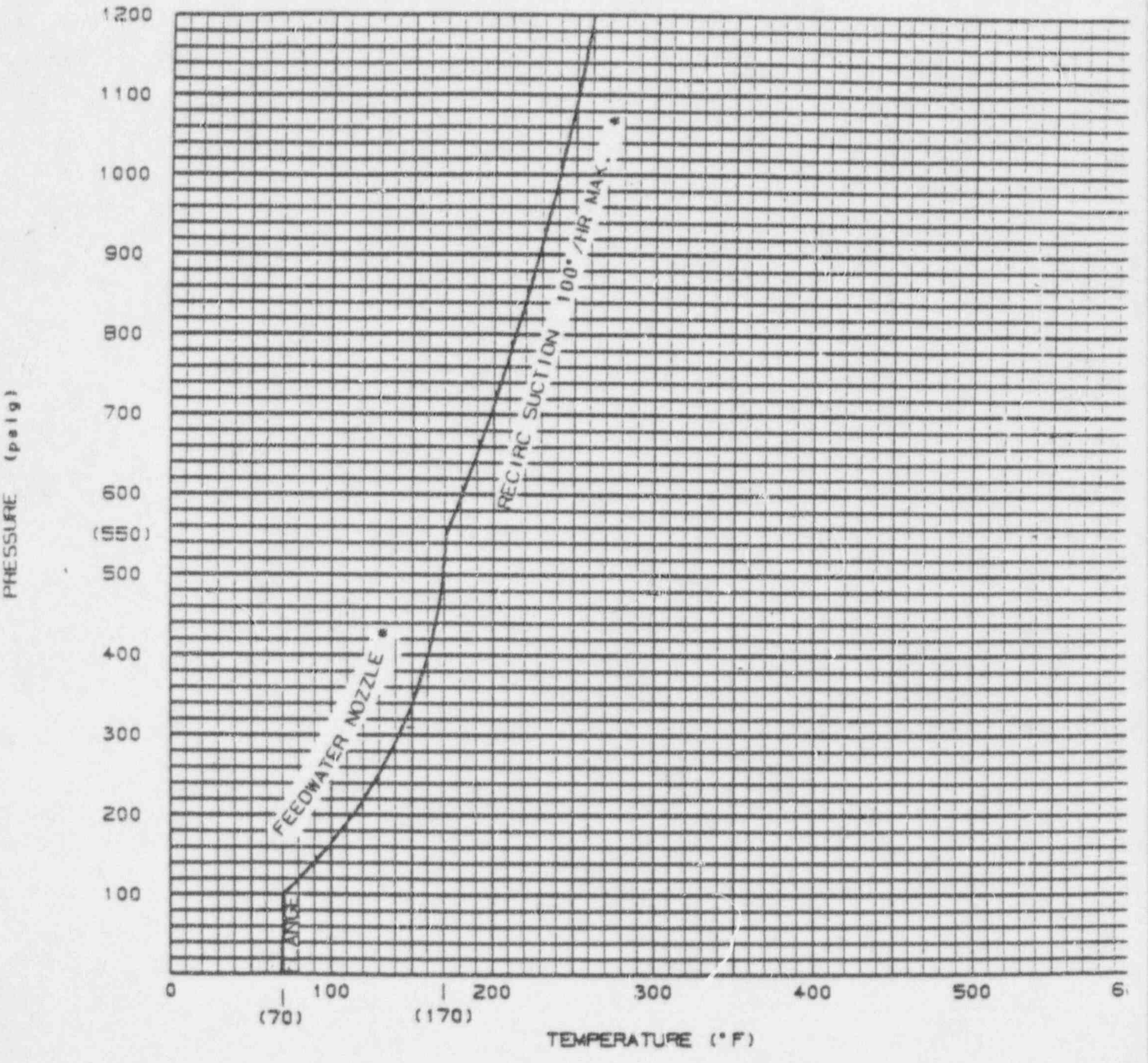
(c) Provides signal to HPCI pump suction valves only.

(d) Alarm only.

(e) Required when ESF equipment is required to be OPERABLE.

FIGURE 3.4.6.1-1
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

- NORMAL OPERATION WITH CORE NOT CRITICAL



BASES:

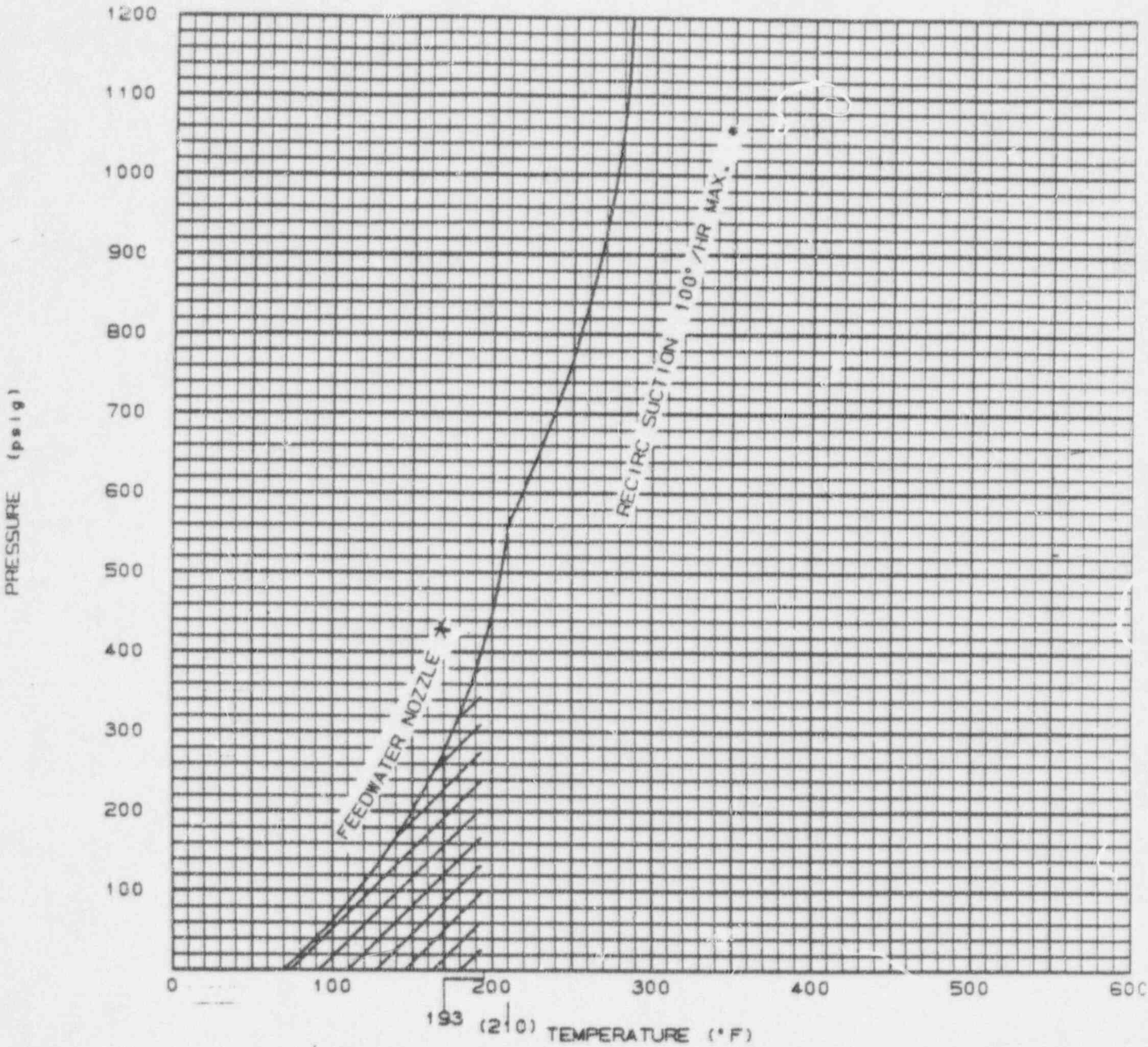
1. FUEL IN REACTOR
2. $\leq$ 16 EFPY
3. 7.1×10^{17} N/CM^2 > 1 MEV
4. RT_{NDT} = 93° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. REG. GUIDE 1.99 REV. 2

NOTES:

1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES

FIGURE 3.4.6.1-2
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

NORMAL OPERATION WITH CORE CRITICAL

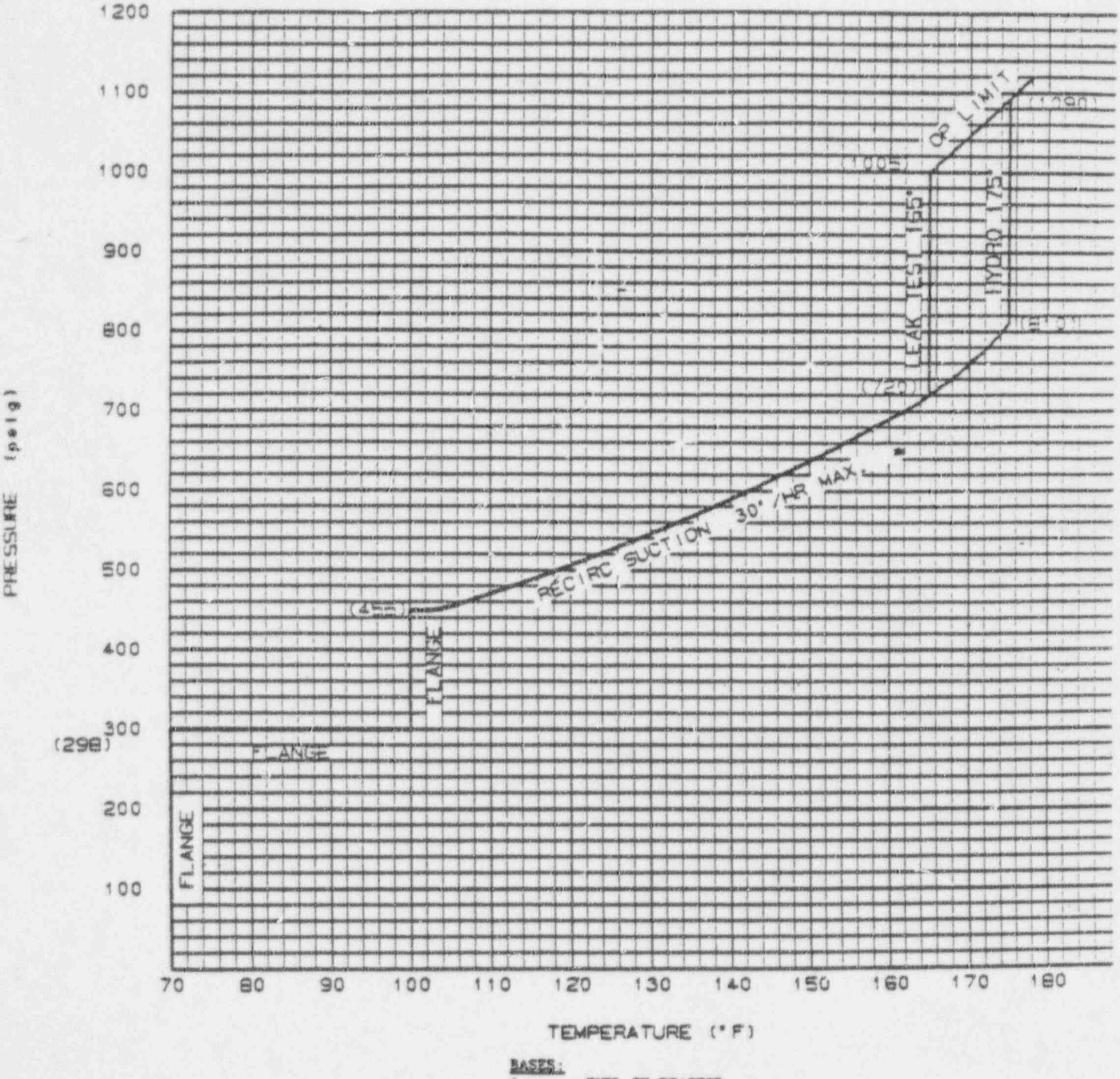


BASES:
1. FUEL IN REACTOR
2. $\leq$ 16 EFPY
3. 7.1×10^{17} N/CM^2 > 1 MEV
4. RT_{NDT} = 93° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. REG. GUIDE 1.99 REV. 2

NOTES:
1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
4. OPERATION IN CROSS-HATCHED AREA PERMITTED ONLY WHEN WATER LEVEL IS WITHIN NORMAL RANGE FOR POWER OPERATION.

FIGURE 3.4.6.1-3a
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS



BASES:

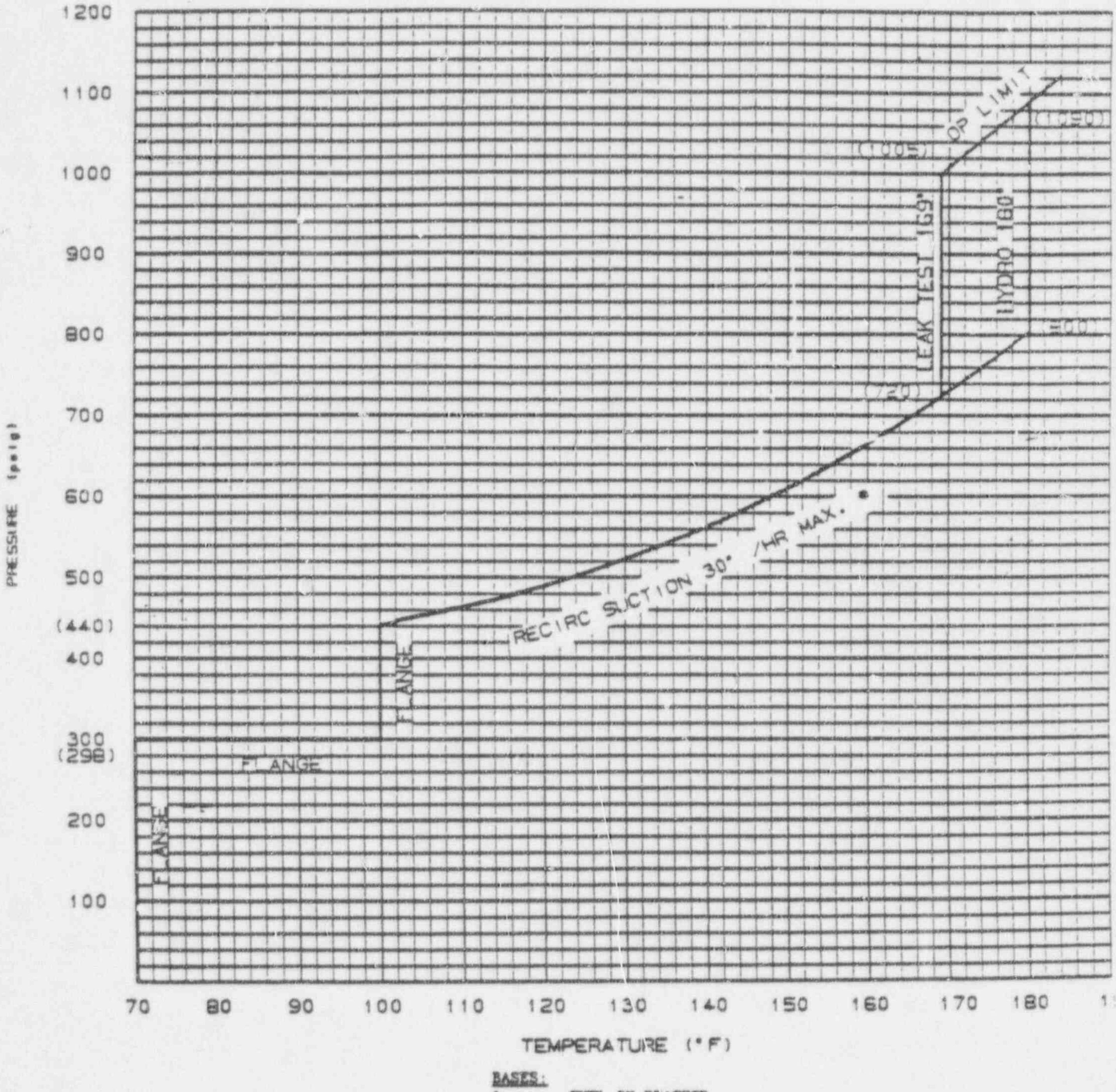
1. FUEL IN REACTOR
2. REACTOR NOT CRITICAL
3. REG. GUIDE 1.99 REV. 2
4. ≤ 8 EFPY
5. 3.5×10^{17} N/CM^2 > 1 MEV
6. RT_{NDT} = 77° (1/4 T)
7. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED

NOTES:

1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
4. OPERATING LIMIT INDICATES TEMPERATURE REQUIRED IF TEST PRESSURE WAS EXCEEDED.

FIGURE 3.4.6.1-3b
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS



BASES:

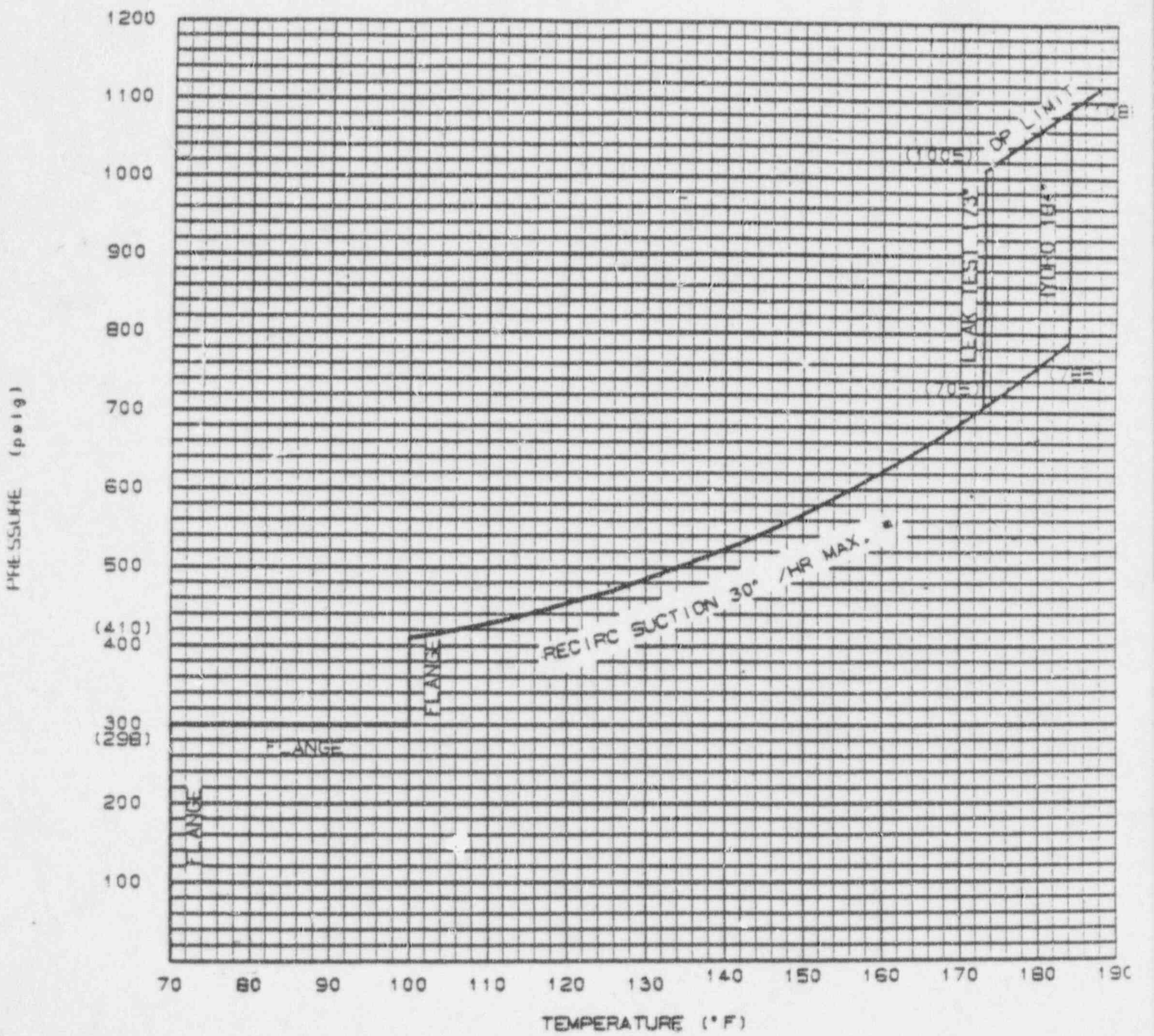
1. FUEL IN REACTOR
2. $\leq$ 10 EFPY
3. 4.4 X 10^{17} N/CM^2 > 1 MEV
4. RT_{NDT} = 82° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. REG. GUIDE 1.99 REV. 2
7. REACTOR NOT CRITICAL

NOTES:

1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
4. OPERATING LIMIT INDICATES TEMPERATURE REQUIRED IF TEST PRESSURE WAS EXCEEDED.

FIGURE 3.4.6.1-3c
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

HYDROSTATIC AND LEAK TESTS



BASES:

1. FUEL IN REACTOR
2. $\leq$ 12 EFPY
3. 5.3×10^{17} N/CM^2 > 1 MEV
4. RT_{NDT} = 88° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. REG. GUIDE 1.99 REV. 2
7. REACTOR NOT CRITICAL

NOTES:

1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
4. OPERATING LIMIT INDICATES TEMPERATURE REQUIRED IF TEST PRESSURE WAS EXCEEDED.

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, $\pm$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.

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.

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.

LOW PRESSURE COOLANT INJECTION SYSTEM

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.

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*Not applicable when two CSS subsystems are OPERABLE per Specification 3.5.3.1.

RETYPED TECH. SPECS.
Updated Thru. Amend. 78

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.

3/4.8 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.

LIMITING CONDITION FOR OPERATION (Continued)

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.

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.

ATTACHMENT 1

Spatial Area Estimation

To estimate the potential spatial area for concern during a hazardous material release, it will be necessary to obtain basic meteorological data and complete Form 1 which is attached to this procedure.

Obtain Meteorological Data:

1. Obtain the following meteorological data from the BNP on-site meteorological monitoring system and complete Form 1:

   Lower wind direction: ______ (deg.) this is the direction from which the wind is blowing.

   Lower wind velocity: ______ (mph)

   If velocity is 0 to 5 mph . . . Light
   If velocity is 5 to 15 mph . . . Moderate
   If velocity is 15 or more . . . Strong

   Stability type: A B C D E F G (circle one)

   If A, B, or C . . . . . Unstable
   If D . . . . Neutral
   If E, F, or G . . . . . Stable

2. If the BNP Control Room cannot provide the required information, contact the National Weather Service (763-8331) at the Wilmington International Airport. Identify yourself and briefly describe the emergency which exists. Ask to speak to the Shift Forecaster On Duty, and request the following information from the **Wilmington Airport Hourly Observation**:

   Wind direction: ______ (deg.) this is the direction from which the wind is blowing.

   Wind velocity: ______ (mph)

   If velocity is 0 to 5 mph . . . Light
   If velocity is 5 to 15 mph . . . Moderate
   If velocity is 15 or more . . . Strong

   Stability Type: Ask the Forecaster if the atmospheric stability from the surface to 500 feet would be classified as:

   (circle one) Unstable
   Neutral
   Stable

3. If neither the BNP Control Room or the National Weather Service can be contacted for weather information, use the following to estimate conditions which exist at the accident site:

Wind Direction: Estimate the direction where north should be. Face that direction and determine from which direction the wind is COMING FROM (to your right will be East.. your left will be West.. from the front will be North.. from the back will be South).

(circle one)	North	Northeast
	East	Southeast
	South	Southwest
	West	Northwest

Wind Velocity: From the following, estimate the category which best describes the wind velocity:

(circle one)

Light	Wind felt on the face; leaves rustle; ordinary wind vanes are moved.
Moderate	Leaves and small twigs on trees in constant motion; wind will extend a light flag.
Strong	Raises dust and loose paper; small branches on trees are moved; small trees in leaf begin to sway.

Stability Class: Use the following to estimate the atmospheric stability affecting the release of gaseous materials.

(circle one)	Stable	Usually occur at night: Clear skies with calm winds and cool surface temperature. Most likely to occur in early morning (2 a.m. until sunrise); also if you observe rising smoke which is suddenly stopped in vertical rise.
	Neutral	Occur during night or day: If skies are cloudy or if there is rain (or rainshowers) then the atmospheric stability is neutral. During evenings from the period of sunset to about 2am; conditions are most likely neutral.
	Unstable	Usually during the daylight hours: Skies must be clear or mostly clear, usually during a warm day with winds moderate or strong. Winds tend to be gusty so that smoke dissipates rapidly.

> **NOTE:** Tables 1 through 3 are used for 25 lbs Cl/sec releases. Assistance from the TSC or engineering may be required for accurate plume spatial extents for other release rates.

4. With the basic meteorological information, an estimate of the spatial extent of a plume can now be made. The following should be done:

Unstable If the atmospheric stability is UNSTABLE, use Table 1 of this procedure. Under the appropriate wind velocity group and the concentration of concern, find the crosswind distance (in feet) and the downwind distance (in feet or miles) where that area of equal concentration may be found.

Example: Moderate wind velocity and you want to find the area of 10 ppm. Use Table 1, look in the middle group under **Moderate** and under the 10 ppm column. At 0.25 miles downwind, the width of the plume of material is 276 feet to the right of the centerline...a total of 552 feet wide at 0.25 miles downwind. The plume is a total of 718 feet wide at 1.25 miles. The area of 10 ppm ended between 1.25 and 1.5 miles downwind of the release.

Neutral If the atmospheric stability is NEUTRAL, use Table 2 of this procedure. Under the appropriate wind velocity group and the concentration of concern, find the crosswind distance (in feet) and the downwind distance (in feet or miles) where that area of equal concentration may be found.

Example: Strong wind velocity and you want to find the area of 10 ppm. Use Table 2, look in the right hand group under **Strong** and under the 10 ppm column. At 0.25 miles downwind, the width of the plume of material is 284 feet to the right of the centerline...a total of 568 feet wide at 0.25 miles downwind. The plume is a total of 968 feet wide at 1.25 miles. The area of 10 ppm ended between 1.75 and 2.0 miles downwind of the release.

Stable If the atmospheric stability is STABLE, use Table 3 of this procedure. Under the appropriate wind velocity group and the concentration of concern, find the crosswind distance (in feet) and the downwind distance (in feet or miles) where that area of equal concentration may be found.

Example: Light wind velocity and you want to find the area of 1000 ppm. Use Table 3, look in the left hand group under **Light** and under the 1000 ppm column. At the widest point of the plume, 1.50 to 1.75 miles from the release point, the plume is only 98 feet to the right of the centerline..... a total of 196 feet wide. However, the 1000 ppm concentration ends between 2.75 and 3.00

miles from the release point. Under Stable conditions and light winds very small concentrations of gaseous material will travel very far downwind provided no other factors act upon the dispersion of the plume.

*** Critical Notes ***

1. Under stable conditions and light wind speeds, gaseous plumes of released material will travel in concentrated form for great distances, provided that no other influences act upon the plume.

   *If the wind does not remain steady in any direction, the plume will not remain concentrated at extended distances, but will disperse the material within a much smaller area. Observation of the wind direction is important not only to estimate the downwind areas of concern, but to assist in the determination of potential concentrations of released material.

   *If there are any close-by structures, large bodies of water or wooded areas to where the release point is located, these influences could locally affect the wind direction and velocities observed. Large objects will direct the wind flow and reduce the velocity. Within a downwind distance ten times the height of a large object, wind direction and velocities will be influenced, thus caution should be exercised in concentration determination.

2. The wind direction reported by meteorologists and the Brunswick Control Room personnel is the wind direction FROM where the wind is coming! Gaseous material released from a source will travel 180° opposite from the wind direction being reported by observation, therefore, in order to have the correct direction for possible actions to be taken, the opposite direction from which the wind is being reported must be used.

3. Meteorological data obtained from the on-site meteorological monitoring system or from the National Weather Service will use compass directions. "Plant North" is rotated 45° to the east of compass north.

T. .E I

CONTINUOUS GAS RELEASE @ 25 LBS CL/SECOND (3.79 M^3/SEC) FOR <u>UNSTABLE</u> ATMOSPHERIC CONDITIONS
CROSSWIND DISTANCE FROM CENTERLINE TO EDGE OF ISOPLETH (FEET) FOR VARIOUS
CONCENTRATIONS OF CL (ppm) AT SEVERAL WIND VELOCITIES

DOWNWIND DISTANCE		LIGHT				MODERATE				STRONG			
MILES	FEET	10	1,000	5,000	10,000	10	1,000	5,000	10,000	10	1,000	5,000	10,000
0.25	1320	335	0	0	0	276	0	0	0	241	0	0	0
0.50	2640	543				407				317			
0.75	3960	701				465				282			
1.00	5280	821				457							
1.25	6600	911				359							
1.50	7920	973											
1.75	9240	1010											
2.00	10560	1019											
2.25	11880	1001											
2.50	13200	949											
2.75	14520	857											
3.00	15840	706											
3.50	18480												
4.00	21120												
4.50	23760												
5.00	26400												
6.00	31680												
7.00	36960												
8.00	42240												
9.00	47250												
10.0	52800												
15.0	79200												
20.0	105600												

TABLE 2

CONTINUOUS GAS RELEASE @ 25 LBS CL/SECOND (3.79 M^3/SEC) FOR <u>NEUTRAL</u> ATMOSPHERIC CONDITIONS
CROSSWIND DISTANCE FROM CENTERLINE TO EDGE OF ISOPLETH (FEET) FOR VARIOUS
CONCENTRATIONS OF CL (ppm) AT SEVERAL WIND VELOCITIES

DOWNWIND DISTANCE		LIGHT				MODERATE				STRONG			
MILES	FEET	10	1,000	5,000	10,000	10	1,000	5,000	10,000	10	1,000	5,000	10,000
0.25	1320	367	0	0	0	341	0	0	0	284	0	0	0
0.50	2640	591				481				414			
0.75	3960	770				595				485			
1.00	5280	919				673				508			
1.25	6600	1047				722				484			
1.50	7920	1156				742				398			
1.75	9240	1251				735				165			
2.00	10560	1332				696							
2.25	11880	1440				618							
2.50	13200	1458				482							
2.75	14520	1504				182							
3.00	15840	1540											
3.50	18480	1582											
4.00	21120	1583											
4.50	23760	1541											
5.00	26400	1450											
6.00	31680	1058											
7.00	36960												
8.00	42240												
9.00	47250												
10.0	52800												
15.0	79200												
20.0	105600												

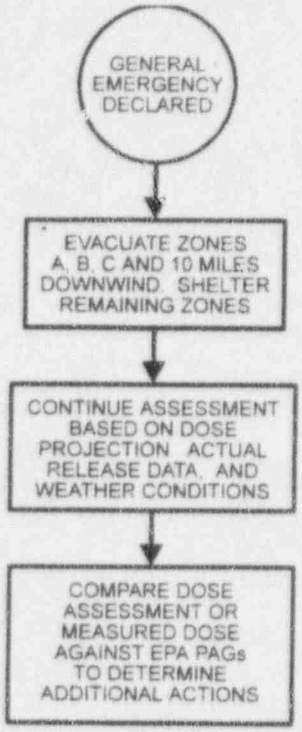
CONTINUOUS GAS RELEASE @ 25 LBS CL/SECOND (3.79 M^3/SEC) FOR <u>STABLE</u> ATMOSPHERIC CONDITIONS
CROSSWIND DISTANCE FROM CENTERLINE TO EDGE OF ISOPLETH (FEET) FOR VARIOUS
CONCENTRATIONS OF CL (ppm) AT SEVERAL WIND VELOCITIES

DOWNWIND DISTANCE		LIGHT				MODERATE				STRONG			
MILES	FEET	10	1,000	5,000	10,000	10	1,000	5,000	10,000	10	1,000	5,000	10,000
0.25	1320	56	38	29	24	51	30	18	8	49	26	8	0
0.50	2640	97	60	38	24	88	42			82	30		
0.75	3960	133	75	37		119	44			111	12		
1.00	5280	166	86	18		147	36			136			
1.25	6600	197	93			173				159			
1.50	7920	226	98			197				180			
1.75	9240	254	98			219				199			
2.00	10560	280	95			240				217			
2.25	11880	305	88			260				234			
2.50	13200	330	75			279				250			
2.75	14520	354	52			297				265			
3.00	15840	377				315				278			
3.50	18480	420				347				304			
4.00	21120	462				377				326			
4.50	23760	502				404				345			
5.00	26400	540				429				362			
6.00	31680	610				474				388			
7.00	36960	676				512				406			
8.00	42240	736				543				414			
9.00	47250	792				569				404			
10.0	52800	845				589							
15.0	79200	1060				607							
20.0	105600	1212				456							

ATTACHMENT 1

Page 1 of 1

PAR Flowchart



ACTIONS FLOW CHART

PROJECTED DOSE (Rem)	RECOMMENDED ACTIONS	COMMENTS
WHOLE BODY (TEDE) <1.0 AND THYROID (CDE) <5.0	NO ADDITIONAL PROTECTIVE ACTION RECOMMENDATIONS REQUIRED	• STATE/COUNTY MAY RECONSIDER PREVIOUS RECOMMENDED PROTECTIVE ACTIONS • CONTINUE MONITORING/ASSESSMENT OF ENVIRONMENTAL RADIATION LEVELS
WHOLE BODY (TEDE) 1 OR ABOVE OR THYROID (CDE) 5 OR ABOVE	RECOMMENDED MANDATORY EVACUATION OF GENERAL PUBLIC FROM THE AFFECTED AREAS	• STATE/COUNTY WILL DETERMINE ACTUAL PROTECTIVE ACTIONS TO BE TAKEN FOR THE GENERAL PUBLIC, TAKING ANY EXISTING CONSTRAINTS OR SPECIAL CONSIDERATIONS INTO ACCOUNT. ALTERNATIVE ACTIONS MAY BE TAKEN BY STATE AND LOCAL AUTHORITIES. • CONTINUE MONITORING/ASSESSMENT OF ENVIRONMENTAL RADIATION LEVELS

ATTACHMENT 2

Page 1 of 1

Evacuation Zones and Time Estimates/10 Mile EPZ Map

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
S[illegible]THEAST (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

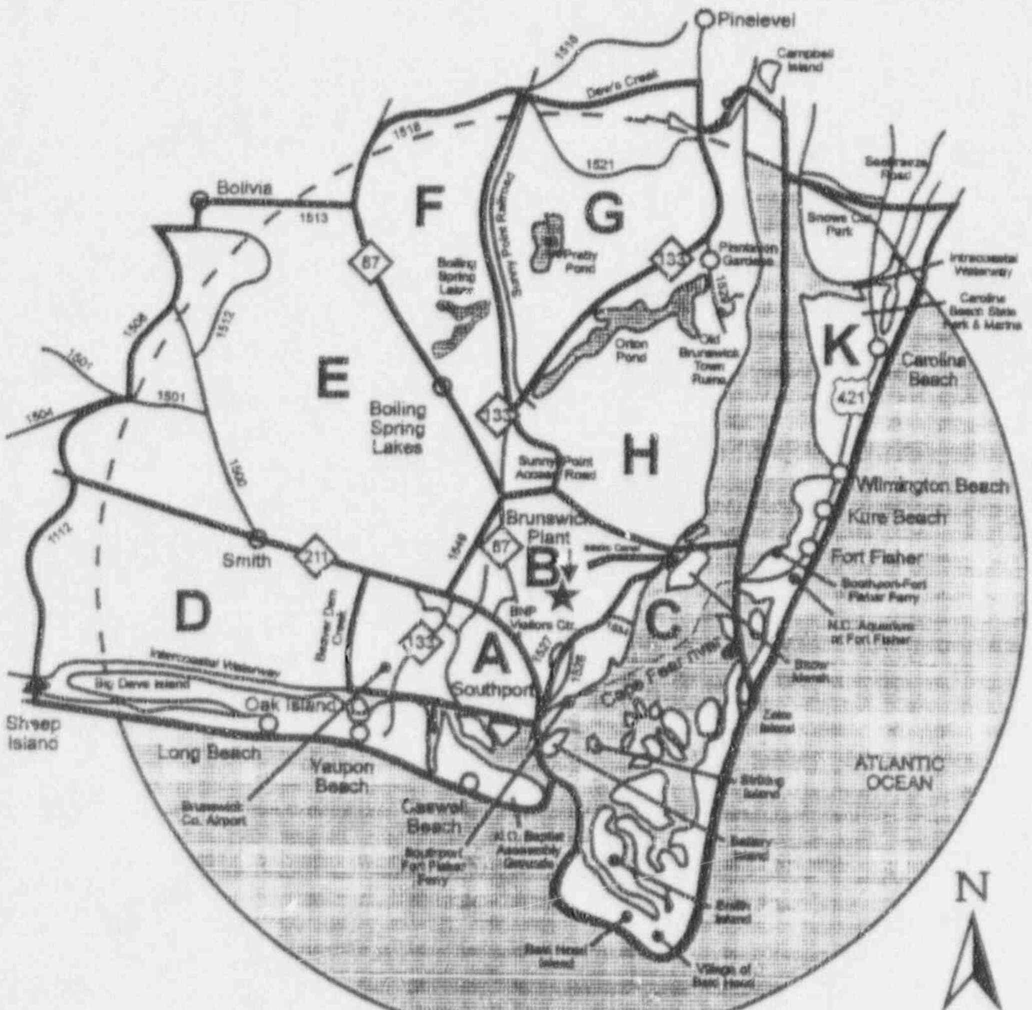


TABLE 1

Reactor Pressure Vs Saturation Temperature

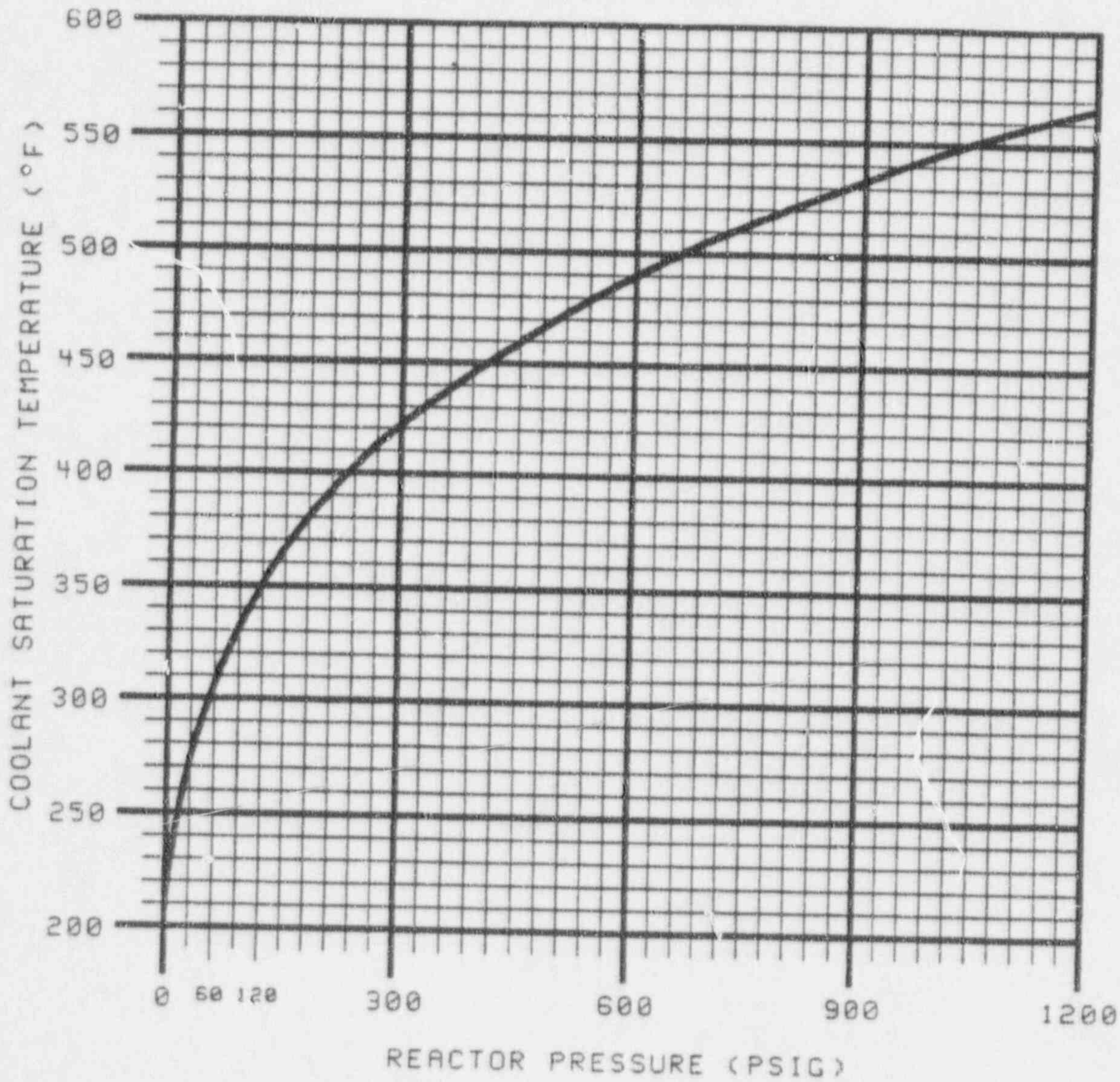
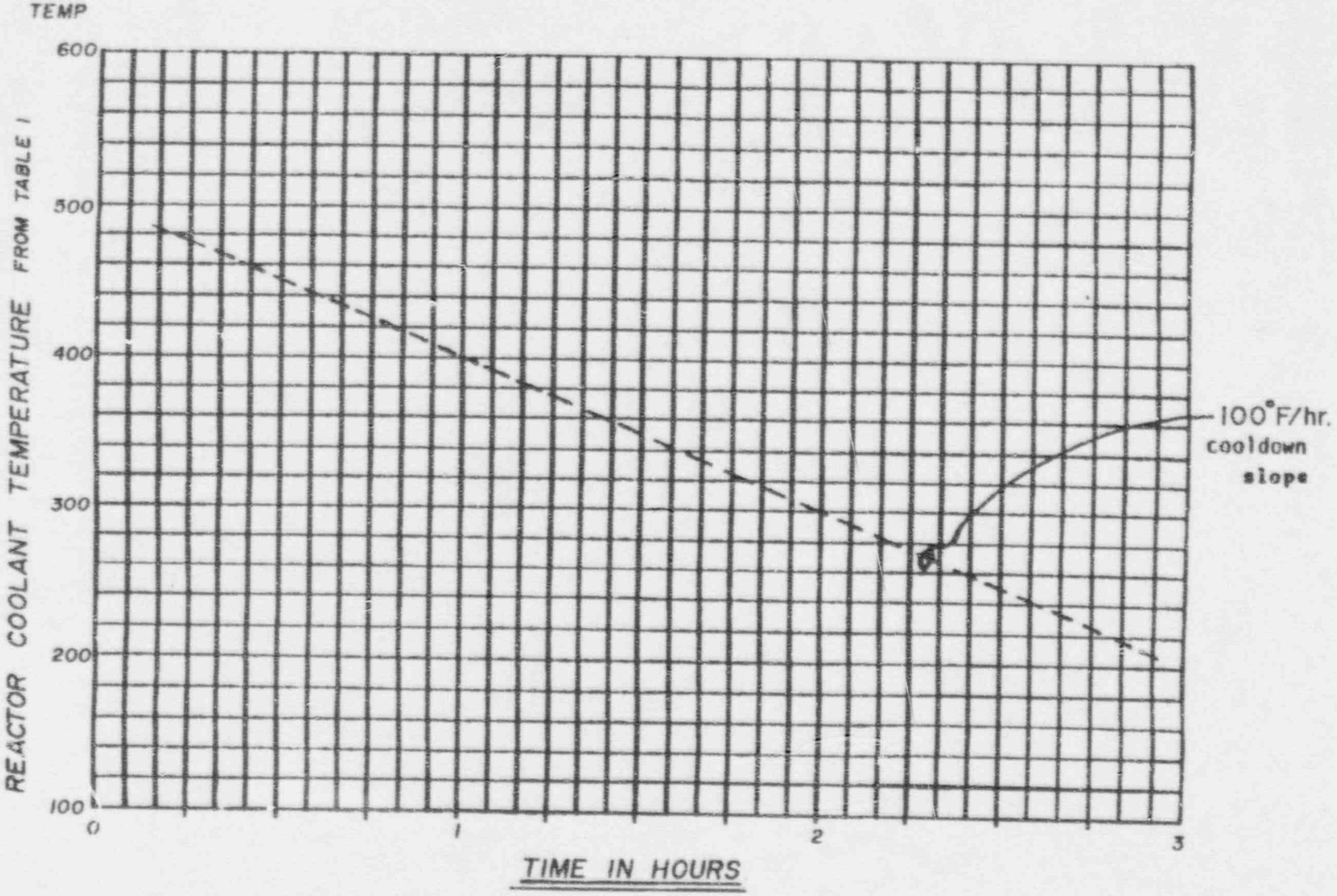


TABLE 2

Reactor Cooldown Plot



Drywell Temperature Calculation Using Remote Shutdown Panel Recorder Inputs

Values are obtained from Recorder CAC-TR-778

Above 70' Elevation

PT 1 ________ x 0.141 = ________ °F

Between 28' and 45' Elevation

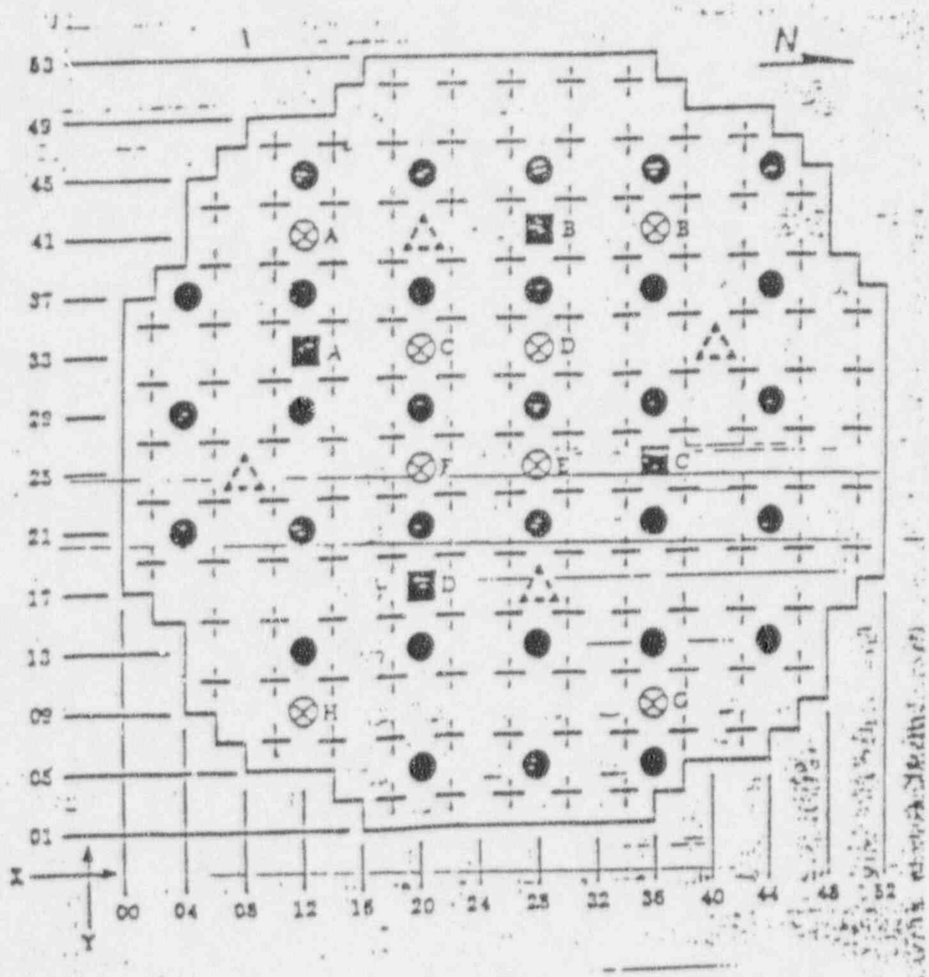
PT 3 ________ x 0.404 = ________ °F

Between 10' and 23' Elevation

PT 4 ________ x 0.455 = ________ °F

Average Drywell Temperature ________ °F
(Sum of 3 Regional Weighted Areas)

FIGURE 4
CORE MAP SHOWING LOCATIONS OF SRM/IRMS



CONTROL RODS (137)

SRM DETECTORS (4)

IRM DETECTORS (8)

SOURCE LOCATIONS (4) (Removed)

IN CORE DETECTOR STRINGS (31) 4 LPRM DETECTORS PER STRING