UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
License No. DPR-71

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated April 2, 1996 (BSEP 96-0123), as supplemented by an earlier submittal dated November 20, 1995 (BSEP 95-0535), and by subsequent submittals dated July 1, 1996 (BSEP 96-0242), July 30, 1996 (BSEP 96-0287), August 7, 1996 (BSEP 96-0300), September 13, 1996 (BSEP 96-0340), September 20, 1996 (BSEP 96-0348), October 1, 1996 (BSEP 96-0362), October 22, 1996 (BSEP 96-0392), October 22, 1996 (BSEP 96-0403), and October 29, 1996 (BSEP 96-0412) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by the revision of paragraph 2.C.(1) and the addition of paragraph 2.L. to the license and by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 183, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented upon completion of refueling outage B111R1.

FOR THE NUCLEAR REGULATORY COMMISSION

Mark Reinhart

Mark Reinhart, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3, 5a, and 5b of License DPR-71
2. Changes to the Technical Specifications

Date of Issuance: November 1, 1996

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2.

(6) Carolina Power and Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report, dated November 22, 1977, as supplemented April 1979, June 11, 1980, December 30, 1986, December 6, 1989, July 28, 1993, and February 10, 1994, respectively, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 183, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

2.I Deleted per Amendment No. 70 dated 5-25-84.

2.J Deleted per Amendment No. 70 dated 5-25-84.

2.K. Augmented Off-Gas System Modifications

The Licensee shall proceed with the necessary design, procurement and construction of the modifications to the Augmented Off-Gas System. By January 15, 1983, the Licensee shall submit proposed Technical Specifications which incorporate effluent limits that reflect the required operation of the augmented off-gas system. By May 31, 1983, the Licensee shall have the augmented off-gas system operable, and the system shall operate in accordance with the referenced Technical Specifications, as issued.

2.L. Power Uprate License Amendment Implementation

The licensee shall complete the following actions as a condition of the approval of the power uprate license amendment (Amendment No. 183):

(1) Control Rod Drive (CRD) System

During initial Unit 1 Cycle 11 start-up testing, if the licensee determines that adequate CRD System cooling and drive flow is not available under power uprate conditions, the licensee shall repair or modify the CRD System, as necessary, to assure that the system will continue to carry out its functions at uprated conditions.

(2) Recirculation Pump Motor Vibration

Perform monitoring of recirculation pump motor vibration during initial Unit 1 Cycle 11 power ascension for uprated power conditions. Vibration and noise shall be evaluated prior to and at uprated conditions to ensure no significant increase in vibration or noise occurs with power uprate.

(3) Fuel Pool Decay Heat Evaluation

The decay heat loads and the decay heat removal systems available for each refueling outage shall be evaluated, and bounding or outage specific analyses shall be used for various refueling sequences. Where a bounding engineering evaluation is in place, a refueling specific assessment shall be made to ensure that the bounding case encompasses the specific refueling sequence. In both cases (i.e., bounding or outage specific evaluations), compliance with design basis assumptions shall be verified.

(4) Equipment Qualification

Environmental qualification of safety-related mechanical equipment with non-metallic components affected by uprated radiation conditions shall be resolved prior to Unit 1 start-up for Cycle 11 operation either by:

(a) Refined radiation calculations (location specific), and/or
(b) Slightly reducing the qualified life, and/or
(c) Assessing the qualification bases by demonstrating qualification based on actual test and materials threshold data while maintaining the regulatory margin, and/or
(d) Assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function.

(5) Human Factors

(a) Classroom Training

Power Uprate Operator Training, including the plant operating parameter changes resulting from power uprate, shall be performed as part of License Operator Retraining (LOR) prior to Unit 1 start-up for Cycle 11 operation.

(b) Simulator Training

Simulator training for power uprate shall be completed prior to Unit 1 start-up for Cycle 11 operation. The simulator training will include the following:

(i) A demonstration of selected transients at the uprated power compared to the non-uprated power, including changes in time to achieve critical points for operator actions.

(ii) The time to meet the conditions to inject boron for a high power ATWS.

(iii) The time to depressurize the reactor on a loss of all high pressure injection (time to achieve conditions requiring emergency depressurization at TAF; non-ATWS).

(c) Simulator Modification

Prior to Unit 1 start-up for Cycle 11, the simulator shall be modified to match the uprated control room, as close as possible, with the exception of the zone coding for the High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System speed indication meters. The HPCI and RCIC speed indication zone coding shall be completed prior to Unit 2 start-up following the implementation of the power uprate license amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by R. C. DeYoung

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. Appendices A, A-Prime, and B - Technical Specifications

Date of Issuance:
September 8, 1976

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

PURGE OR PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the containment.

RATED THERMAL POWER

RATED THERMAL POWER shall be total reactor core heat transfer rate to the reactor coolant of 2558 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

# TABLE 2.2.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|---|
| 1. | Intermediate Range Monitor, Neutron Flux - High[a] | ≤ 120 divisions of full scale | ≤ 120 divisions of full scale |
| 2. | Average Power Range Monitor | | |
| | a. Neutron Flux - High, 15%[b] | ≤ 15% of RATED THERMAL POWER | ≤ 15% of RATED THERMAL POWER |
| | b. Flow-Biased Simulated Thermal Power - High[c][d] | ≤ (0.66W + 59.6%) with a maximum ≤ 113.6% of RATED THERMAL POWER | ≤ (0.66W + 61%) with a maximum ≤ 115.3% of RATED THERMAL POWER |
| | c. Fixed Neutron Flux - High[d] | ≤ 116.3% of RATED THERMAL POWER | ≤ 118% of RATED THERMAL POWER |
| 3. | Reactor Vessel Steam Dome Pressure - High | ≤ 1067.9 psig | ≤ 1070 psig |
| 4. | Reactor Vessel Water Level - Low, Level 1 | ≥ +153.2 inches[g] | ≥ +153 inches[g] |
| 5. | Main Steam Line Isolation Valve - Closure[e] | ≤ 10% closed | ≤ 10% closed |
| 6. | (Deleted) | | |
| 7. | Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| 8. | Scram Discharge Volume Water Level - High | ≤ 109 gallons | ≤ 109 gallons |
| 9. | Turbine Stop Valve - Closure[f] | ≤ 10% closed | ≤ 10% closed |
| 10. | Turbine Control Valve Fast Closure, Control Oil Pressure - Low[f] | ≥ 500 psig | ≥ 500 psig |

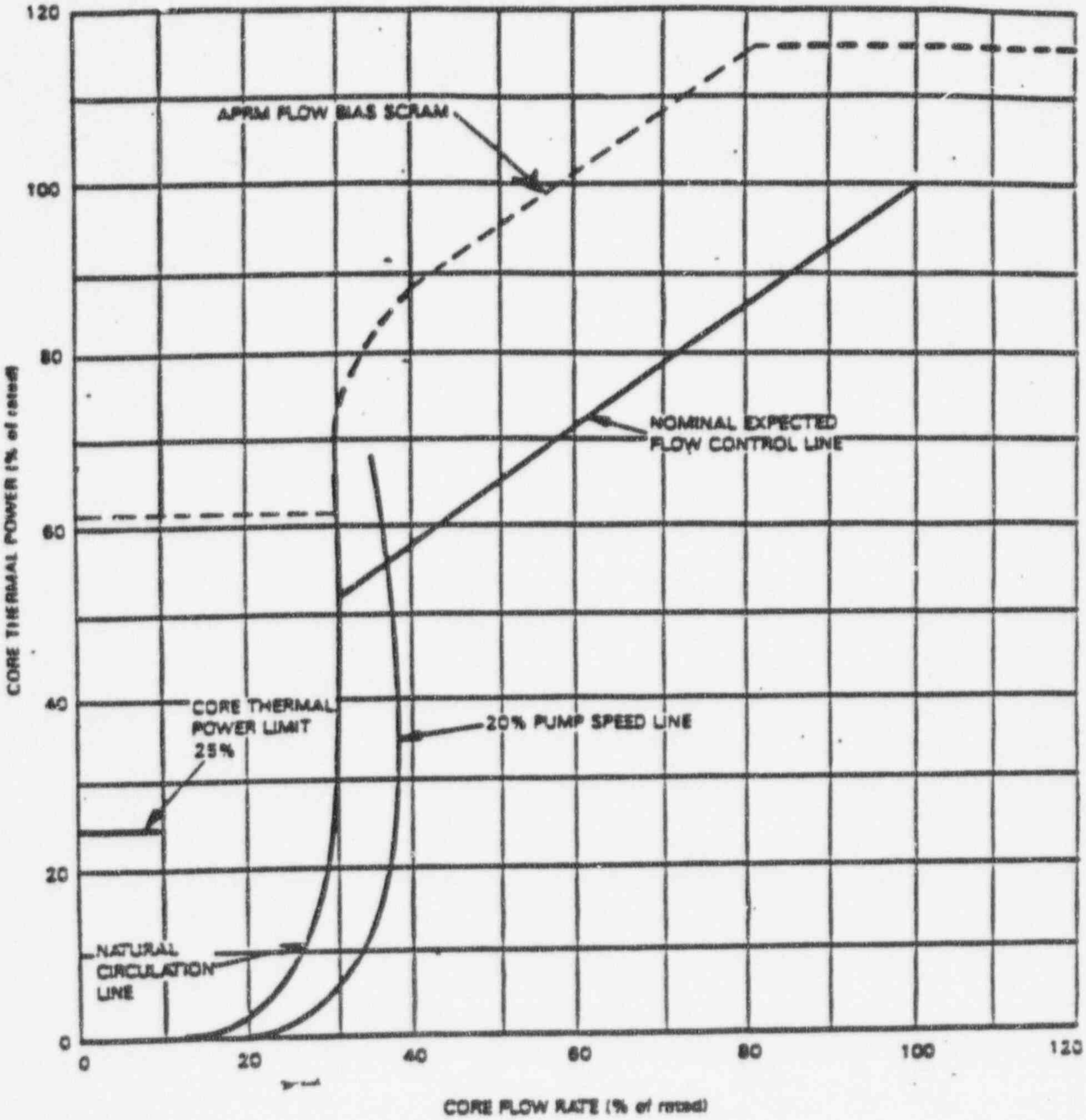



Figure 2.2.1-1. APRM Flow Bias Scram Relationship to Normal Operating Conditions

2. Average Power Range Monitor (Continued)

be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the RUN position.

The APRM flux scram trip in the RUN mode consists of a flow biased simulated thermal power (STP) scram setpoint and a fixed neutron flux scram setpoint. The APRM flow biased neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal, e.g., STP, that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions.

The APRM flow biased simulated thermal power scram trip setting at full recirculation flow is adjustable up to the nominal trip setpoint of 113.6% of RATED THERMAL POWER. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 116.3% fixed neutron flux scram trip. The lower flow biased scram setpoint therefore decreases the severity, ΔCPR, of a slow thermal transient and allows lower operating limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the fuel cycle.

The APRM fixed neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow biased simulated thermal power scram.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High Pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating, will also tend to increase the power of the reactor by compressing voids, thus adding reactivity. The trip will quickly reduce the neutron flux counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This setpoint is effective at low power/flow conditions when the turbine stop valve closure is bypassed. For a turbine trip under these conditions, the transient analysis indicates a considerable margin to the thermal hydraulic limit.

## TABLE 3.3.2-2

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. PRIMARY CONTAINMENT ISOLATION | | |
| a. Reactor Vessel Water Level - | | |
| 1. Low, Level 1 | $\geq$ + 153.2 inches[(a)] | $\geq$ + 153 inches[(a)] |
| 2. Low, Level 3 | $\geq$ + 14.1 inches[(a)] | $\geq$ + 13 inches[(a)] |
| b. Drywell Pressure - High | $\leq$ 2 psig | $\leq$ 2 psig |
| c. Main Steam Line | | |
| 1. (Deleted) | | |
| 2. Pressure - Low | $\geq$ 825 psig | $\geq$ 825 psig |
| 3. Flow - High | $\leq$ 137% of rated flow | $\leq$ 138% of rated flow |
| d. Main Steam Line Tunnel Temperature - High | $\leq$ 200°F | $\leq$ 200°F |
| e. Condenser Vacuum - Low | $\geq$ 7.6 inches Hg vacuum | $\geq$ 7.5 inches Hg vacuum |
| f. Turbine Building Area Temperature - High | $\leq$ 200°F | $\leq$ 200°F |
| g. Main Stack Radiation - High | (b) | (b) |
| h. Reactor Building Exhaust Radiation - High | $\leq$ 11 mr/hr | $\leq$ 11 mr/hr |

## TABLE 3.3.2-2 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 2. SECONDARY CONTAINMENT ISOLATION | | |
| a. Reactor Building Exhaust Radiation - High | ≤ 11 mr/hr | ≤ 11 mr/hr |
| b. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| c. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[a] | ≥ + 103 inches[a] |
| 3. REACTOR WATER CLEANUP SYSTEM ISOLATION | | |
| a. Δ Flow - High | ≤ 73 gal/min | ≤ 73 gal/min |
| b. Area Temperature - High | ≤ 150°F | ≤ 150°F |
| c. Area Ventilation Δ Temperature - High | ≤ 50°F | ≤ 50°F |
| d. SLCS Initiation | NA | NA |
| e. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[a] | ≥ + 103 inches[a] |
| f. Δ Flow - High - Time Delay | ≤ 30 minutes | ≤ 30 minutes |
| g. Piping Outside RWCU Rooms Area Temperature - High | ≤ 120°F | ≤ 120°F |

## TABLE 3.3.2-2 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 4. CORE STANDBY COOLING SYSTEMS ISOLATION | | |
| a. High Pressure Coolant Injection System Isolation | | |
| 1. HPCI Steam Line Flow - High | $\leq$ 272% of rated flow | $\leq$ 275% of rated flow |
| 2. HPCI Steam Line Flow - High Time Delay Relay | $3 \leq t \leq 7$ seconds | $3 \leq t \leq 12$ seconds |
| 3. HPCI Steam Supply Pressure - Low | $\geq$ 106.6 psig | $\geq$ 104 psig |
| 4. HPCI Steam Line Tunnel Temperature - High | $\leq$ 200°F | $\leq$ 200°F |
| 5. Bus Power Monitor | NA | NA |
| 6. HPCI Turbine Exhaust Diaphragm Pressure - High | $\leq$ 8.5 psig | $\leq$ 9 psig |
| 7. HPCI Steam Line Ambient Temperature - High | $\leq$ 200°F | $\leq$ 200°F |
| 8. HPCI Steam Line Area $\Delta$ Temperature - High | $\leq$ 50°F | $\leq$ 50°F |
| 9. HPCI Equipment Area Temperature - High | $\leq$ 175°F | $\leq$ 175°F |
| 10. Drywell Pressure - High | $\leq$ 2 psig | $\leq$ 2 psig |

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued) | | |
| b. Reactor Core Isolation Cooling System Isolation | | |
| 1. RCIC Steam Line Flow - High | ≤ 272% of rated flow | ≤ 275% of rated flow |
| 2. RCIC Steam Line Flow - High Time Delay Relay | 3 ≤ t ≤ 7 seconds | 3 ≤ t ≤ 12 seconds |
| 3. RCIC Steam Supply Pressure - Low | ≥ 55.6 psig | ≥ 53 psig |
| 4. RCIC Steam Line Tunnel Temperature - High | ≤ 175°F | ≤ 175°F |
| 5. Bus Power Monitor | NA | NA |
| 6. RCIC Turbine Exhaust Diaphragm Pressure - High | ≤ 5.5 psig | ≤ 6 psig |
| 7. RCIC Steam Line Ambient Temperature - High | ≤ 200°F | ≤ 200°F |
| 8. RCIC Steam Line Area Δ Temperature - High | ≤ 50°F | ≤ 50°F |
| 9. RCIC Equipment Room Ambient Temperature - High | ≤ 175°F | ≤ 175°F |
| 10. RCIC Equipment Room Δ Temperature - High | ≤ 50°F | ≤ 50°F |
| 11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay | ≤ 30 minutes | ≤ 30 minutes |
| 12. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 5. SHUTDOWN COOLING SYSTEM ISOLATION | | |
| a. Reactor Vessel Water Level - Low Level 1 | $\geq$ 153.2 inches[a] | $\geq$ 153 inches[a] |
| b. Reactor Steam Dome Pressure - High | $\leq$ 130.8 psig | $\leq$ 137 psig |

(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

(b) Establish alarm/trip setpoints per the methodology contained in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

## TABLE 3.3.3-2

## EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. CORE SPRAY SYSTEM | | |
| a. Reactor Vessel Water Level - Low, Level 3 | $\geq$ + 14.1 inches[(b)] | $\geq$ + 13 inches[(b)] |
| b. Reactor Steam Dome Pressure - Low | $\geq$ 406.7 psig | $\geq$ 404 psig |
| c. Drywell Pressure - High | $\leq$ 2 psig | $\leq$ 2 psig |
| d. Time Delay-Relay | $14 \leq t \leq 16$ secs | $14 \leq t \leq 16$ secs |
| e. Bus Power Monitor | NA | NA |
| 2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM | | |
| a. Drywell Pressure - High | $\leq$ 2 psig | $\leq$ 2 psig |
| b. Reactor Vessel Water Level - Low, Level 3 | $\geq$ + 14.1 inches[(b)] | $\geq$ + 13 inches[(b)] |
| c. Reactor Vessel Shroud Level | $\geq$ - 53 inches[(b)] | $\geq$ - 53 inches[(b)] |
| d. Reactor Steam Dome Pressure - Low | | |
| 1. RHR Pump Start and LCPI Valve Actuation | $\geq$ 406.7 psig | $\geq$ 404 psig |
| 2 Recirculation Pump Discharge Valve Actuation | $\geq$ 306.7 psig | $\geq$ 304 psig |
| e. RHR Pump Start - Time Delay Relay | $9 \leq t \leq 11$ seconds | $9 \leq t \leq 11$ seconds |
| f. Bus Power Monitor | NA | NA |

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 3. HIGH PRESSURE COOLANT INJECTION SYSTEM | | |
| a. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[b] | ≥ + 103 inches[b] |
| b. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| c. Condensate Storage Tank Level - Low | ≥ 23 feet 4 inches | ≥ 23 feet 4 inches |
| d. Suppression Chamber Water Level - High | ≤ -2 feet[c] | ≤ -2 feet[c] |
| e. Bus Power Monitor | NA | NA |
| 4. AUTOMATIC DEPRESSURIZATION SYSTEM | | |
| a. ADS Inhibit Switch | NA | NA |
| b. Reactor Vessel Water Level - Low, Level 3 | ≥ + 14.1 inches[b] | ≥ + 13 inches[b] |
| c. Reactor Vessel Water Level - Low, Level 1 | ≥ + 153.2 inches[b] | ≥ + 153 inches[b] |
| d. ADS Timer | ≤ 83 seconds | ≤ 108 seconds |
| e. Core Spray Pump Discharge Pressure - High | ≥ 112.1 psig | ≥ 102 psig |
| f. RHR (LPCI MODE) Pump Discharge Pressure - High | ≥ 111.1 psig | ≥ 102 psig |
| g. Bus Power Monitor | NA | NA |

TABLE 3.3.4-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. APRM | | |
| a. Upscale (Flow Biased) | ≤ (0.66W + 54.6%)[a] with a maximum of ≤ 109.3% of RATED THERMAL POWER | ≤ (0.66W + 56%)[a] with a maximum of ≤ 111% of RATED THERMAL POWER |
| b. Inoperative | NA | NA |
| c. Downscale | ≥ 3/125 of full scale | ≥ 3/125 of full scale |
| d. Upscale (Fixed) | ≤ 12% of RATED THERMAL POWER | ≤ 12% of RATED THERMAL POWER |
| 2. ROD BLOCK MONITOR | | |
| a. Upscale | As specified in the CORE OPERATING LIMITS REPORT | As specified in the CORE OPERATING LIMITS REPORT |
| b. Inoperative | NA | NA |
| c. Downscale | ≥ 94/125 of full scale | NA |
| 3. SOURCE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | ≤ 1 x $10^5$ cps | ≤ 1 x $10^5$ cps |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 3 cps | ≥ 3 cps |
| 4. INTERMEDIATE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | ≤ 108/125 of full scale | ≤ 108/125 of full scale |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 3/125 of full scale | ≥ 3/125 of full scale |
| 5. SCRAM DISCHARGE VOLUME | | |
| a. Water Level High | ≤ 73 gallons | ≤ 73 gallons |

(a) Where W is the fraction of rated recirculation loop flow in percent.

## TABLE 3.3.6.1-2

## ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. Reactor Vessel Water Level - Low, Level 2 | $\geq$ + 104.1 inches[a] | $\geq$ + 103 inches[a] |
| 2. Reactor Vessel Pressure - High | $\leq$ 1137.8 psig | $\leq$ 1143 psig |

[a] Vessel water levels refer to REFERENCE LEVEL ZERO.

## TABLE 3.3.7-2

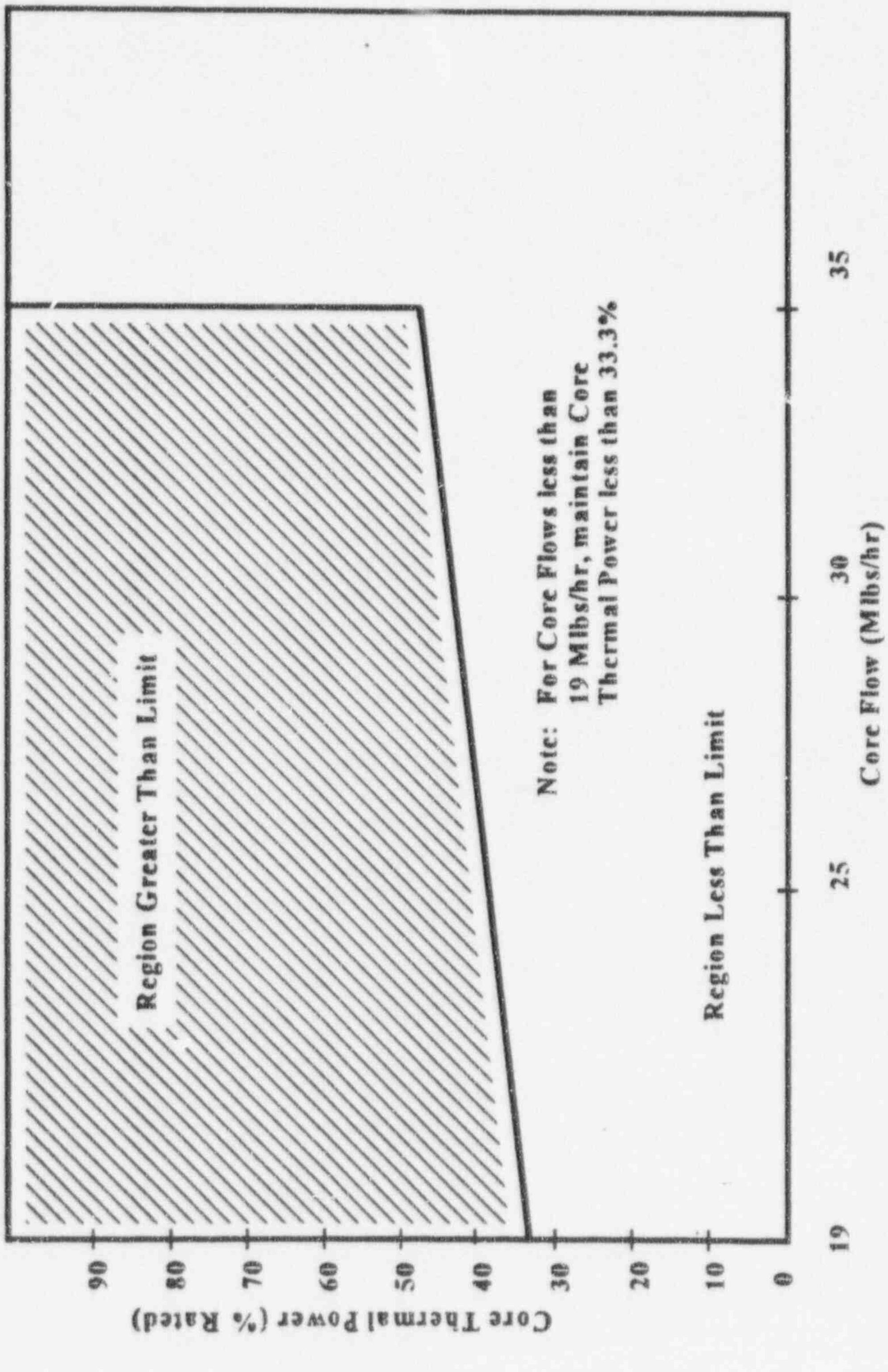
## REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[a] | ≥ + 103 inches[a] |
| 2. Reactor Vessel Water Level - High | ≤ + 206.8 inches[a] | ≤ + 207 inches[a] |
| 3. Condensate Storage Tank Level - Low | ≥ 23 feet 0 inches | ≥ 23 feet 0 inches |

(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

FIGURE 3.4.1.1-1

THERMAL POWER LIMITATIONS



## 3/4.4.2 SAFETY/RELIEF VALVES

### LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 10 reactor coolant system safety/relief valves shall be OPERABLE with lift settings of the required valves within ± 3% of the following values.*

4 Safety-relief valves @ 1130 psig.
4 Safety-relief valves @ 1140 psig.
3 Safety-relief valves @ 1150 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

### SURVEILLANCE REQUIREMENTS

4.4.2 The safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE in accordance with the Surveillance Requirements of Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valves at normal operating temperature and pressure.

## REACTOR STEAM DOME

### LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1045 psig.

APPLICABILITY: CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

### SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

* Not applicable during anticipated transients, reactor isolation, or reactor trip.

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 ≥ 1025 psig when steam is being supplied to the turbine at 1025, +20, -80, 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 ≥ 165 psig when steam is being supplied to the turbine at 165, ± 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.

PLANT SYSTEMS

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

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3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 113 psig.

ACTION:

With the RCIC system inoperable, operation may continue and the provisions of Specifications 3.0.4 are not applicable provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 113 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the highpoint vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 400 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1025 + 20, - 80 psig.*

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after reactor steam pressure is adequate to perform the test.

These specifications are based on the guidance of General Electric SIL #380, Rev. 1, 2-10-84.

3/4.4.2 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code Section III for the pressure vessel and ANSI B31.1, 1975, Code for the reactor coolant system piping.

The GE analysis (GE-NE-B21-00565-03) provided as part of the Power Uprate project assumed one (1) SRV out of service for the ATWS transient and two (2) SRVs out of service for the limiting over pressure transient. The LCO and Action Statement reflects the limiting complement of SRVs which is the 10 assumed in the ATWS analysis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Monitoring leakage at eight hour intervals is in conformance with the 12/21/89 NRC SER for GL 88-01.

3/4.4.4 CHEMISTRY

The reactor water chemistry limits are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus, the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present.

3/4.4.4 CHEMISTRY (continued)

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity outside the limits, additional samples must be examined to ensure that the chlorides are not exceeding the limits.

## 3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the calculated pressure of 49 psig during primary system blowdown from full operating pressure.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1045 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the design pressure of 62 psig. Maximum water volume of 89,600 $ft^3$ results in a downcomer submergence of 3'4" and the minimum volume of 87,600 $ft^3$ results in a submergence approximately four inches less. The Monticello tests were run with a submerged length of three feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

When it is necessary to make the suppression chamber inoperable, this shall only be done as provided in Specification 3.5.3.3.

Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown, which is below the temperature 170°F used for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps; thus, there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated April 2, 1996 (BSEP 96-0123), as supplemented by an earlier submittal dated November 20, 1995 (BSEP 95-0535), and by subsequent submittals dated July 1, 1996 (BSEP 96-0242), July 30, 1996 (BSEP 96-0287), August 7, 1996 (BSEP 96-0300), September 13, 1996 (BSEP 96-0340), September 20, 1996 (BSEP 96-0348), October 1, 1996 (BSEP 96-0362), October 22, 1996 (BSEP 96-0392), October 22, 1996 (BSEP 96-0403), and October 29, 1996 (BSEP 96-0412) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by the revision of paragraph 2.C.(1) and the addition of paragraph 2.I. to the license and by changes to the Technical Specifications as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than the start-up of Unit 2 following refueling outage B213R1.

FOR THE NUCLEAR REGULATORY COMMISSION

Mark Reinhart

Mark Reinhart, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
(1) Pages 3, 4a, and 4b of License DPR-62
(2) Changes to the Technical
Specifications

Date of Issuance: November 1, 1996

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source, and special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Part 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2.

(6) Carolina Power and Light Company shall implement and maintain in effect all provision of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated November 22, 1977, as supplemented April 1979, June 11, 1980, December 30, 1986, December 6, 1989, July 28, 1993 and February 10, 1994 respectively, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

the licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts (thermal).

2.F Deleted per Amendment No. 98 dated 5-25-84.

2.G Deleted per Amendment No. 98 dated 5-25-84.

2.H Augmented Off-Gas System Modifications

The licensee shall proceed with the necessary design, procurement and construction of the modifications to the augmented off-gas system. By July 15, 1983, the Licensee shall submit proposed Technical Specifications which incorporate effluent limits that reflect the required operation of the augmented off-gas system. Within two months following the extended outage scheduled to begin in March 1984, the Licensee shall have the augmented off-gas system operable, and the system shall operate in accordance with the referenced Technical Specifications, as issued.

2.I. Power Uprate License Amendment Implementation

The licensee shall complete the following actions as a condition of the approval of the power uprate license amendment (Amendment No. 214):

(1) Control Rod Drive (CRD) System

During initial start-up testing following implementation of the power uprate license amendment, the licensee shall determine that adequate CRD System cooling and drive flow is available under power uprate conditions. If adequate CRD System cooling and drive flow is not available under power uprate conditions, the licensee shall repair or modify the CRD System, as necessary, to assure that the system will continue to carry out its functions at uprated conditions.

(2) Recirculation Pump Motor Vibration

Upon implementation of the power uprate license amendment, perform monitoring of recirculation pump motor vibration during the initial power ascension for uprated power conditions. Vibration and noise shall be evaluated prior to and at uprated conditions to ensure no significant increase in vibration or noise occurs with power uprate.

(3) Fuel Pool Decay Heat Evaluation

The decay heat loads and the decay heat removal systems available for each refueling outage shall be evaluated, and bounding or outage specific analyses shall be used for various refueling sequences. Where a bounding engineering evaluation is in place, a refueling specific assessment shall be made to ensure that the bounding case encompasses the specific refueling sequence. In both cases (i.e., bounding or outage specific evaluations), compliance with design basis assumptions shall be verified.

(4) Equipment Qualification

Environmental qualification of safety-related mechanical equipment with non-metallic components affected by uprated radiation conditions shall be resolved prior to Unit 1 start-up for Cycle 11 operation either by:

(a) Refined radiation calculations (location specific), and/or
(b) Slightly reducing the qualified life, and/or
(c) Assessing the qualification bases by demonstrating qualification based on actual test and materials threshold data while maintaining the regulatory margin, and/or
(d) Assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function.

(5) Human Factors

(a) Simulator Modification

Prior to the initial start-up following implementation of the power uprate license amendment, the simulator shall be modified to match the uprated Unit 2 control room, as close as possible.

Original signed by:

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendices A & B -
Technical Specifications

Date of Issuance: Dec. 27, 1974

ATTACHMENT TO LICENSE AMENDMENT NO. 214

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

| Removed page | Inserted page |
|---|---|
| 1-6 | 1-6 |
| 2-4 | 2-4 |
| 2-6 | 2-6 |
| B 2-5 | B 2-5 |
| 3/4 3-18 | 3/4 3-18 |
| 3/4 3-19 | 3/4 3-19 |
| 3/4 3-20 | 3/4 3-20 |
| 3/4 3-21 | 3/4 3-21 |
| 3/4 3-22 | 3/4 3-22 |
| 3/4 3-39 | 3/4 3-39 |
| 3/4 3-40 | 3/4 3-40 |
| 3/4 3-50 | 3/4 3-50 |
| 3/4 3-91 | 3/4 3-91 |
| 3/4 3-102 | 3/4 3-102 |
| 3/4 4-1b | 3/4 4-1b |
| 3/4 4-4 | 3/4 4-4 |
| 3/4 4-21 | 3/4 4-21 |
| 3/4 5-2 | 3/4 5-2 |
| 3/4 7-7 | 3/4 7-7 |
| B 3/4 4-2 | B 3/4 4-2 |
| -- | B 3/4 4-2a |
| B 3/4 6-3 | B 3/4 6-3 |

PRIMARY CONTAINMENT INTEGRITY (Continued)

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

PURGE OR PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the containment.

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2558 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

## TABLE 2.2.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Intermediate Range Monitor, Neutron Flux - High[a] | ≤ 120 divisions of full scale | ≤ 120 divisions of full scale |
| 2. Average Power Range Monitor | | |
| a. Neutron Flux - High, 15%[b] | ≤ 15% of RATED THERMAL POWER | ≤ 15% of RATED THERMAL POWER |
| b. Flow-Biased Simulated Thermal Power - High[c][d] | ≤ (0.66W + 59.6%) with a maximum ≤ 113.6% of RATED THERMAL POWER | ≤ (0.66W + 61%) with a maximum ≤ 115.3% of RATED THERMAL POWER |
| c. Fixed Neutron Flux - High[d] | ≤ 116.3% of RATED THERMAL POWER | ≤ 118% of RATED THERMAL POWER |
| 3. Reactor Vessel Steam Dome Pressure - High | ≤ 1067.9 psig | ≤ 1070 psig |
| 4. Reactor Vessel Water Level - Low, Level 1 | ≥ +153.2 inches[g] | ≥ +153 inches[g] |
| 5. Main Steam Line Isolation Valve - Closure[e] | ≤ 10% closed | ≤ 10% closed |
| 6. (Deleted) | | |
| 7. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| 8. Scram Discharge Volume Water Level - High | ≤ 109 gallons | ≤ 109 gallons |
| 9. Turbine Stop Valve - Closure[f] | ≤ 10% closed | ≤ 10% closed |
| 10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low[f] | ≥ 500 psig | ≥ 500 psig |

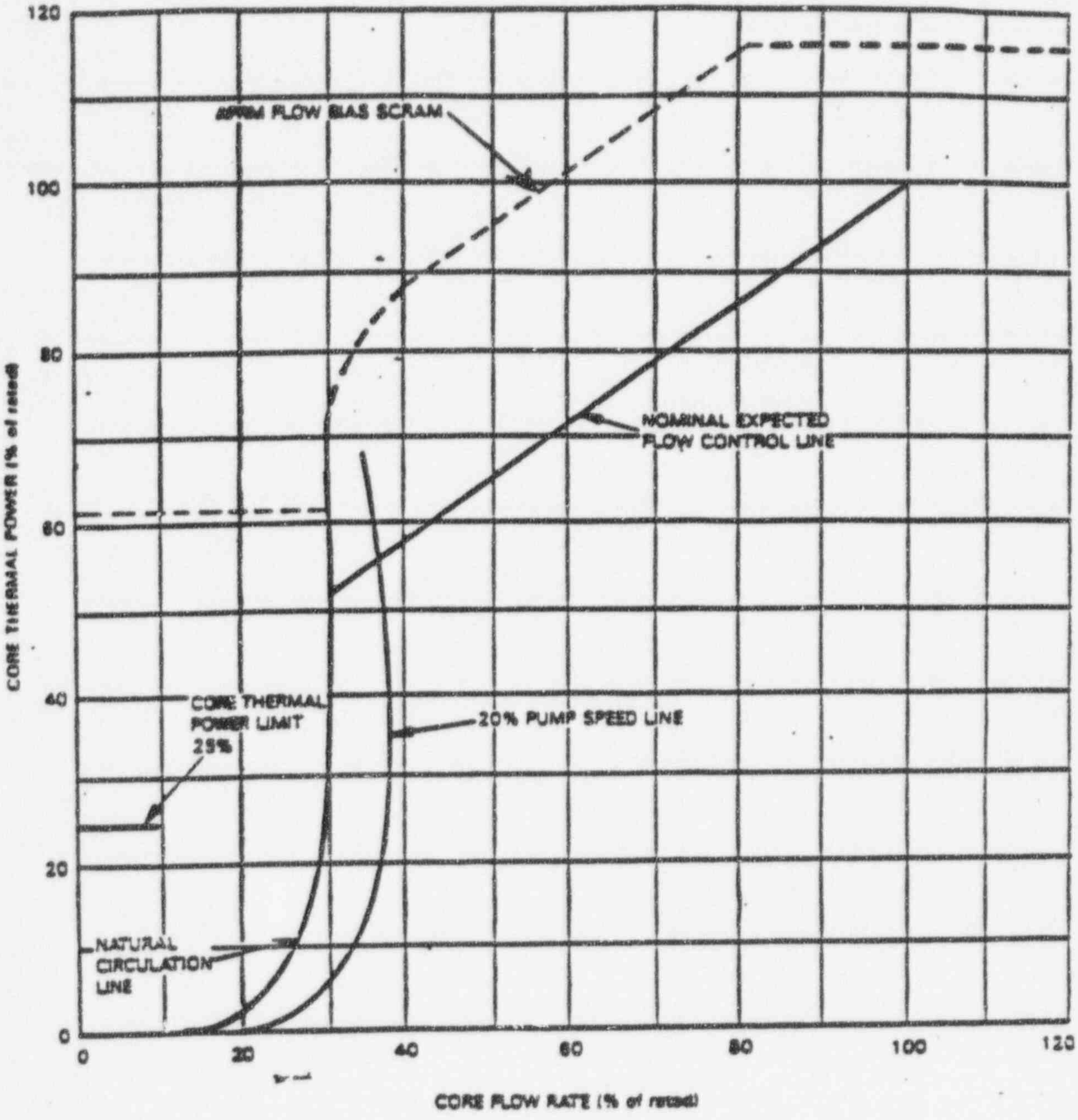


Figure 2.2.1-1. APRM Flow Bias Scram Relationship to Normal Operating Conditions

2. Average Power Range Monitor (Continued)

minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the RUN position.

The APRM flux scram trip in RUN mode consists of a flow biased simulated thermal power (STP) scram setpoint and a fixed neutron flux scram setpoint. The APRM flow biased neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal, e.g., STP, that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions.

The APRM flow biased simulated thermal power scram trip setting at full recirculation flow is adjustable up to the nominal trip setpoint of 113.6% of RATED THERMAL POWER. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 116.3% fixed neutron flux scram trip. The lower flow biased scram setpoint therefore decreases the severity, ΔCPR, of a slow thermal transient and allows lower operating limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the fuel cycle.

The APRM fixed neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow biased simulated thermal power scram.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High Pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids, thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase by decreasing heat generation. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This setpoint is effective at low power/flow conditions when the turbine stop valve closure is bypassed. For a turbine trip under these conditions, the transient analysis indicates a considerable margin to the thermal hydraulic limit.

## TABLE 3.3.2-2

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. PRIMARY CONTAINMENT ISOLATION | | |
| a. Reactor Vessel Water Level - | | |
| 1. Low, Level 1 | ≥ + 153.2 inches[a] | ≥ + 153 inches[a] |
| 2. Low, Level 3 | ≥ + 14.1 inches[a] | ≥ + 13 inches[a] |
| b. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| c. Main Steam Line | | |
| 1. (Deleted) | | |
| 2. Pressure - Low | ≥ 825 psig | ≥ 825 psig |
| 3. Flow - High | ≤ 137% of rated flow | ≤ 138% of rated flow |
| 4. Flow - High | ≤ 30% of rated flow | ≤ 32% of rated flow |
| d. Main Steam Line Tunnel Temperature - High | ≤ 200°F | ≤ 200°F |
| e. Condenser Vacuum - Low | ≥ 7.6 inches Hg vacuum | ≥ 7.5 inches Hg vacuum |
| f. Turbine Building Area Temperature - High | ≤ 200°F | ≤ 200°F |
| g. Main Stack Radiation - High | (b) | (b) |
| h. Reactor Building Exhaust Radiation - High | ≤ 11 mr/hr | ≤ 11 mr/hr |

## TABLE 3.3.2-2 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 2. SECONDARY CONTAINMENT ISOLATION | | |
| a. Reactor Building Exhaust Radiation - High | ≤ 11 mr/hr | ≤ 11 mr/hr |
| b. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| c. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[a] | ≥ + 103 inches[a] |
| 3. REACTOR WATER CLEANUP SYSTEM ISOLATION | | |
| a. Δ Flow - High | ≤ 73 gal/min | ≤ 73 gal/min |
| b. Area Temperature - High | ≤ 150°F | ≤ 150°F |
| c. Area Ventilation Temperature Δ Temp - High | ≤ 50°F | ≤ 50°F |
| d. SLCS Initiation | NA | NA |
| e. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[a] | ≥ + 103 inches[a] |
| f. Δ Flow - High - Time Delay | ≤ 30 minutes | ≤ 30 minutes |
| g. Piping Outside RWCU Rooms Area Temperature - High | ≤ 120°F | ≤ 120° |

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 4. CORE STANDBY COOLING SYSTEMS ISOLATION | | |
| a. High Pressure Coolant Injection System Isolation | | |
| 1. HPCI Steam Line Flow - High | ≤ 272% of rated flow | ≤ 275% of rated flow |
| 2. HPCI Steam Line Flow - High Time Delay Relay | 3 ≤ t ≤ 7 seconds | 3 ≤ t ≤ 12 seconds |
| 3. HPCI Steam Supply Pressure - Low | ≥ 106.6 psig | ≥ 104 psig |
| 4. HPCI Steam Line Tunnel Temperature - High | ≤ 200°F | ≤ 200°F |
| 5. Bus Power Monitor | NA | NA |
| 6. HPCI Turbine Exhaust Diaphragm Pressure - High | ≤ 8.5 psig | ≤ 9 psig |
| 7. HPCI Steam Line Ambient Temperature - High | ≤ 200°F | ≤ 200°F |
| 8. HPCI Steam Line Area Δ Temperature - High | ≤ 50°F | ≤ 50°F |
| 9. HPCI Equipment Area Temperature - High | ≤ 175°F | ≤ 175°F |
| 10. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 4. CORE STANDBY COOLING SYSTEMS ISOLATION (Continued) | | |
| b. Reactor Core Isolation Cooling System Isolation | | |
| 1. RCIC Steam Line Flow - High | $\leq$ 272% of rated flow | $\leq$ 275% of rated flow |
| 2. RCIC Steam Line Flow - High Time Delay Relay | $3 \leq t \leq 7$ seconds | $3 \leq t \leq 12$ seconds |
| 3. RCIC Steam Supply Pressure - Low | $\geq$ 55.6 psig | $\geq$ 53 psig |
| 4. RCIC Steam Line Tunnel Temperature - High | $\leq$ 175°F | $\leq$ 175°F |
| 5. Bus Power Monitor | NA | NA |
| 6. RCIC Turbine Exhaust Diaphragm Pressure - High | $\leq$ 5.5 psig | $\leq$ 6 psig |
| 7. RCIC Steam Line Ambient Temperature - High | $\leq$ 200°F | $\leq$ 200°F |
| 8. RCIC Steam Line Area $\Delta$ Temperature - High | $\leq$ 50°F | $\leq$ 50°F |
| 9. RCIC Equipment Room Ambient Temperature - High | $\leq$ 175°F | $\leq$ 175°F |
| 10. RCIC Equipment Room $\Delta$ Temperature - High | $\leq$ 50°F | $\leq$ 50°F |
| 11. RCIC Steam Line Tunnel Temperature - High Time Delay Relay | $\leq$ 30 minutes | $\leq$ 30 minutes |
| 12. Drywell Pressure - High | $\leq$ 2 psig | $\leq$ 2 psig |

## TABLE 3.3.2-2 (Continued)

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 5. SHUTDOWN COOLING SYSTEM ISOLATION | | |
| a. Reactor Vessel Water Level - Low Level 1 | $\geq$ 153.2 inches[a] | $\geq$ 153 inches[a] |
| b. Reactor Steam Dome Pressure - High | $\leq$ 130.8 psig | $\leq$ 137 psig |

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(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

(b) Establish alarm/trip setpoints per the methodology contained in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

## TABLE 3.3.3-2

## EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. CORE SPRAY SYSTEM | | |
| a. Reactor Vessel Water Level - Low, Level 3 | ≥ + 14.1 inches[b] | ≥ + 13 inches[b] |
| b. Reactor Steam Dome Pressure - Low | ≥ 406.7 psig | ≥ 404 psig |
| c. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| d. Time Delay-Relay | $14 \leq t \leq 16$ secs | $14 \leq t \leq 16$ secs |
| e. Bus Power Monitor | NA | NA |
| 2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM | | |
| a. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| b. Reactor Vessel Water Level - Low, Level 3 | ≥ + 14.1 inches[b] | ≥ + 13 inches[b] |
| c. Reactor Vessel Shroud Level | ≥ - 53 inches[b] | ≥ - 53 inches[b] |
| d. Reactor Steam Dome Pressure - Low | | |
| 1. RHR Pump Start and LCPI Valve Actuation | ≥ 406.7 psig | ≥ 404 psig |
| 2. Recirculation Pump Discharge Valve Actuation | ≥ 306.7 psig | ≥ 304 psig |
| e. RHR Pump Start - Time Delay Relay | $9 \leq t \leq 11$ seconds | $9 \leq t \leq 11$ seconds |
| f. Bus Power Monitor | NA | NA |

## TABLE 3.3.3-2 (Continued)

## EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 3. HIGH PRESSURE COOLANT INJECTION SYSTEM | | |
| a. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[b] | ≥ + 103 inches[b] |
| b. Drywell Pressure - High | ≤ 2 psig | ≤ 2 psig |
| c. Condensate Storage Tank Level - Low | ≥ 23 feet 4 inches | ≥ 23 feet 4 inches |
| d. Suppression Chamber Water Level - High | ≤ -2 feet[c] | ≤ -2 feet[c] |
| e. Bus Power Monitor | NA | NA |
| 4. AUTOMATIC DEPRESSURIZATION SYSTEM | | |
| a. ADS Inhibit Switch | NA | NA |
| b. Reactor Vessel Water Level - Low, Level 3 | ≥ + 14.1 inches[b] | ≥ + 13 inches[b] |
| c. Reactor Vessel Water Level - Low, Level 1 | ≥ + 153.2 inches[b] | ≥ + 153 inches[b] |
| d. ADS Timer | ≤ 83 seconds | ≤ 108 seconds |
| e. Core Spray Pump Discharge Pressure - High | ≥ 112.1 psig | ≥ 102 psig |
| f. RHR (LPCI MODE) Pump Discharge Pressure - High | ≥ 111.1 psig | ≥ 102 psig |
| g. Bus Power Monitor | NA | NA |

## TABLE 3.3.4-2

## CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. APRM | | |
| a. Upscale (Flow Biased) | ≤ (0.66W + 54.6%)[a] with a maximum of ≤ 109.3% of RATED THERMAL POWER | ≤ (0.66W + 56%)[a] with a maximum of ≤ 111% of RATED THERMAL POWER |
| b. Inoperative | NA | NA |
| c. Downscale | ≥ 3/125 of full scale | ≥ 3/125 of full scale |
| d. Upscale (Fixed) | ≤ 12% of RATED THERMAL POWER | ≤ 12% of RATED THERMAL POWER |
| 2. ROD BLOCK MONITOR | | |
| a. Upscale | As specified in the CORE OPERATING LIMITS REPORT | As specified in the CORE OPERATING LIMITS REPORT |
| b. Inoperative | NA | NA |
| c. Downscale | ≥ 94/125 of full scale | NA |
| 3. SOURCE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | ≤ 1 x $10^5$ cps | ≤ 1 x $10^5$ cps |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 3 cps | ≥ 3 cps |
| 4. INTERMEDIATE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | ≤ 108/125 of full scale | ≤ 108/125 of full scale |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 3/125 of full scale | ≥ 3/125 of full scale |
| 5. SCRAM DISCHARGE VOLUME | | |
| a. Water Level High | ≤ 73 gallons | ≤ 73 gallons |

(a) Where W is the fraction of rated recirculation loop flow in percent.

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. Reactor Vessel Water Level - Low, Level 2 | $\geq$ + 104.1 inches[a] | $\geq$ + 103 inches[a] |
| 2. Reactor Vessel Pressure - High | $\leq$ 1137.8 psig | $\leq$ 1143 psig |

[a] Vessel water levels refer to REFERENCE LEVEL ZERO.

## TABLE 3.3.7-2

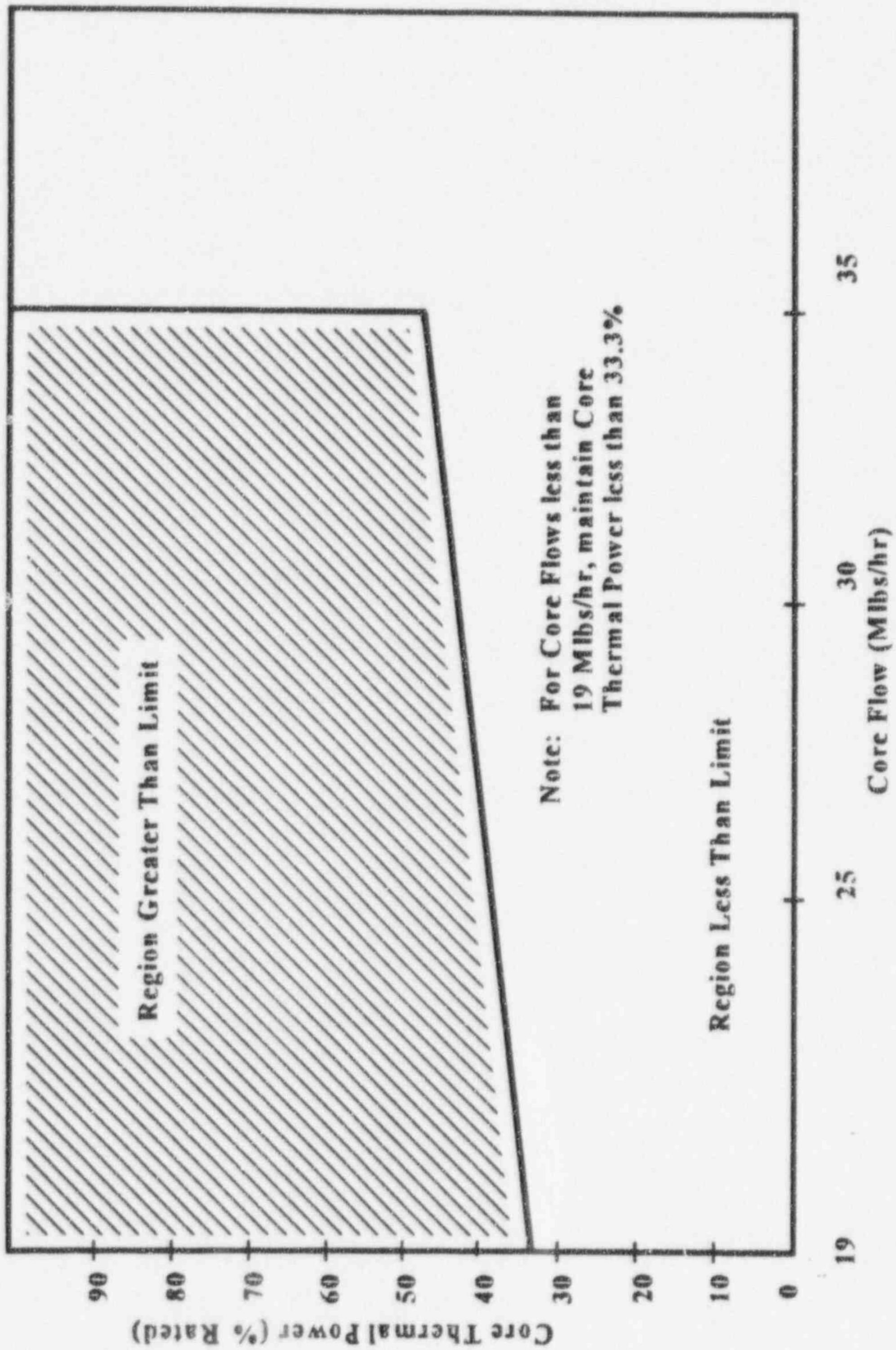
## REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. Reactor Vessel Water Level - Low, Level 2 | ≥ + 104.1 inches[a] | ≥ + 103 inches[a] |
| 2. Reactor Vessel Water Level - High | ≤ +206.8 inches[a] | ≤ +207 inches[a] |
| 3. Condensate Storage Tank Level - Low | ≥ 23 feet 0 inches | ≥ 23 feet 0 inches |

(a) Vessel water levels refer to REFERENCE LEVEL ZERO.

FIGURE 3.4.1.1-1

THERMAL POWER LIMITATIONS



## 3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of 10 reactor coolant system safety/relief valves shall be OPERABLE with lift settings of the required valves within ± 3% of the following values.*

4 Safety-relief valves @ 1130 psig.
4 Safety-relief valves @ 1140 psig.
3 Safety-relief valves @ 1150 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE in accordance with the Surveillance Requirements of Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valves at normal operating temperature and pressure.

## REACTOR STEAM DOME

### LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1045 psig.

APPLICABILITY: CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

### SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

* Not applicable during anticipated transients, reactor isolation, or reactor trip.

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 ≥ 1025 psig when steam is being supplied to the turbine at 1025, +20, -80, 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 ≥ 165 psig when steam is being supplied to the turbine at 165, ± 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.

## 3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

### LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 113 psig.

ACTION:

With the RCIC system inoperable, operation may continue and the provisions of Specifications 3.0.4 are not applicable provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 113 psig within the following 24 hours.

### SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the highpoint vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 400 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1025 + 20, - 80 psig.*

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 24 hours after reactor steam pressure is adequate to perform the test.

These specifications are based on the guidance of General Electric SIL #380, Rev. 1, 2-10-84.

## 3/4.4.2 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code Section III for the pressure vessel and ANSI B31.1, 1975, Code for the reactor coolant system piping.

The GE analysis (GE-NE-B21-00565-03) provided as part of the Power Uprate project assumed one (1) SRV out of service for the ATWS transient and two (2) SRVs out of service for the limiting over pressure transient. The LCO and Action Statement reflects the limiting complement of SRVs which is the 10 assumed in the ATWS analysis.

## 3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

### 3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

### 3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Monitoring leakage at eight hour intervals is in conformance with the 12/21/89 NRC SER for GL 88-01.

## 3/4.4.4 CHEMISTRY

The reactor water chemistry limits are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus, the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present.

3/4.4.4 CHEMISTRY (continued)

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity outside the limits, additional samples must be examined to ensure that the chlorides are not exceeding the limits.

## 3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the calculated pressure of 49 psig during primary system blowdown from full operating pressure.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1045 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the design pressure of 62 psig. Maximum water volume of 89,600 $ft^3$ results in a downcomer submergence of 3'4" and the minimum volume of 87,600 $ft^3$ results in a submergence approximately four inches less. The Monticello tests were run with a submerged length of three feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown test during the Humboldt Bay and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

When it is necessary to make the suppression chamber inoperable, this shall only be done as provided in Specification 3.5.3.3.

Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown, which is below the temperature 170°F used for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps; thus, there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.