



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

January 2, 1985

Docket No. 50-395

Mr. O. W. Dixon, Jr.  
Vice President Nuclear Operations  
South Carolina Electric & Gas Company  
P.O. Box 764 (Mail Code 167)  
Columbia, South Carolina 29218

Dear Mr. Dixon:

Subject: Issuance of Amendment No. 35 to Facility Operating  
License NPF-12 Virgil C. Summer Nuclear Station,  
Unit No. 1

The Nuclear Regulatory Commission has issued Amendment No.35 to Facility Operating License NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1 located in Fairfield County, South Carolina. This amendment is in response to your letter dated February 22, 1984.

The amendment modifies the Technical Specification reporting requirements to be in accordance with new regulation 10 CFR 50.73. The amendment is effective as of its date of issuance.

A copy of the related safety evaluation supporting Amendment No.35 to Facility Operating License NPF-12 is enclosed.

Sincerely,

Elinor G. Adensam, Chief  
Licensing Branch No. 4  
Division of Licensing

Enclosures:

1. Amendment No. 35
2. Safety Evaluation

cc w/enclosure:  
See next page

DESIGNATED ORIGINAL

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SUMMER (Amendment & Order)

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

SOUTH CAROLINA ELECTRIC & GAS COMPANY  
SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35  
License No. NPF-12

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment to the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility) Facility Operating License No. NPF-12 filed by the South Carolina Electric & Gas Company acting for itself and South Carolina Public Service Authority (the licensees), dated February 22, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CFR Chapter I;
  - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;
  - E. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and paragraph 2.C(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 35, are hereby incorporated into this license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Elinor G. Adensam, Chief  
Licensing Branch No. 4  
Division of Licensing

Enclosure:  
Technical Specification Change

Date of Issuance: January 2, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are also provided to maintain document completeness.

<u>Amended</u> <u>Pages</u>	<u>Overleaf</u> <u>Pages</u>
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## DEFINITIONS

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### QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

### RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 Mwt.

### REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

### REPORTABLE EVENT

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

### SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

### SLAVE RELAY TEST

1.29 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

### SOLIDIFICATION

1.30 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a uniformly distributed, monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

### SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

## DEFINITIONS

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### STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

- a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,
- b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

### THERMAL POWER

1.33 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

### TRIP ACTUATING DEVICE OPERATIONAL TEST

1.34 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

### UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

### VENTILATION EXHAUST TREATMENT SYSTEM

1.36 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

### VENTING

1.37 VENTING 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 not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### BASES

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#### 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the reactor trip setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1,  $Z + R + S \leq TA$ , the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

## LIMITING SAFETY SYSTEM SETTINGS

### BASES

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#### REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Protection System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Protection System which monitors numerous system variables, therefore, providing protection system functional diversity. The Reactor Protection System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive reactor system cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

#### Manual Reactor Trip

The Reactor Protection System includes manual reactor trip capability.

#### Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a high and low range trip setting. The low setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the high setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The low setpoint trip may be manually blocked above P-10 (a power level of approximately 10 percent of RATED THERMAL POWER) and is automatically reinstated below the P-10 setpoint.

#### Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power, a rod drop accident of a single or multiple rods could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBR's will be greater than 1.30.

#### Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an

## POWER DISTRIBUTION LIMITS

### SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2  $F_{xy}$  shall be evaluated to determine if  $F_Q(Z)$  is within its limit by:

- a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.
- b. Increasing the measured  $F_{xy}$  component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.
- c. Comparing the  $F_{xy}$  computed ( $F_{xy}^C$ ) obtained in b, above to:
  1. The  $F_{xy}$  limits for RATED THERMAL POWER ( $F_{xy}^{RTP}$ ) for the appropriate measured core planes given in e. and f. below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)]$$

where  $F_{xy}^L$  is the limit for fractional THERMAL POWER operation expressed as a function of  $F_{xy}^{RTP}$  and P is the fraction of RATED THERMAL POWER at which  $F_{xy}$  was measured.

- d. Remeasuring  $F_{xy}$  according to the following schedule:

1. When  $F_{xy}^C$  is greater than the  $F_{xy}^{RTP}$  limit for the appropriate measured core plane but less than the  $F_{xy}^L$  relationship, additional power distribution maps shall be taken and  $F_{xy}^C$  compared to  $F_{xy}^{RTP}$  and  $F_{xy}^L$ :
  - a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which  $F_{xy}^C$  was last determined, or
  - b) At least once per 31 EFPD, whichever occurs first.

## POWER DISTRIBUTION LIMITS

### SURVEILLANCE REQUIREMENTS (Continued)

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2. When the  $F_{xy}^C$  is less than or equal to the  $F_{xy}^{RTP}$  limit for the appropriate measured core plane, additional power distribution maps shall be taken and  $F_{xy}^C$  compared to  $F_{xy}^{RTP}$  and  $F_{xy}^L$  at least once per 31 EFPD.
  - e. The  $F_{xy}$  limits for RATED THERMAL POWER ( $F_{xy}^{RTP}$ ) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.11.
  - f. The  $F_{xy}$  limits of e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:
    1. Lower core region from 0 to 15%, inclusive.
    2. Upper core region from 85 to 100%, inclusive.
    3. Grid plane regions at  $17.8 \pm 2\%$ ,  $32.1 \pm 2\%$ ,  $46.4 \pm 2\%$ ,  $60.6 \pm 2\%$  and  $74.9 \pm 2\%$ , inclusive. (17 x 17 fuel elements).
    4. Core plane regions within  $\pm 2\%$  of core height ( $\pm 2.88$  inches) about the bank demand position of the bank "D" control rods.
  - g. With  $F_{xy}^C$  exceeding  $F_{xy}^L$  the effects of  $F_{xy}$  on  $F_Q(Z)$  shall be evaluated to determine if  $F_Q(Z)$  is within its limits.
- 4.2.2.3 When  $F_Q(Z)$  is measured for other than  $F_{xy}$  determinations, an overall measured  $F_Q(Z)$  shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

## INSTRUMENTATION

### RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

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3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

#### ACTION:

- a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.
- b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally if this condition prevails for more than 30 days, in the next semiannual effluent report, explain why this condition was not corrected in a timely manner.
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

## INSTRUMENTATION

### RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

---

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

- a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.
- b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally if this condition prevails for more than 30 days, in the next semiannual effluent report, explain why this condition was not corrected in a timely manner.
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

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4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	<u>INSTRUMENT</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABILITY</u>	<u>ACTION</u>
1.	WASTE GAS HOLDUP SYSTEM			
a.	Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release RM-A10 or RM-A3	1	*	38
2.	WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a.	Oxygen Monitor	2	**	44
b.	Hydrogen Monitor	1	**	42
3.	MAIN PLANT VENT EXHAUST SYSTEM			
a.	Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release from Waste Gas Holdup System RM-A3	1	*	40
b.	Iodine Sampler	1	*	43
c.	Particulate Sampler	1	*	43
d.	Flow Rate Measuring Device	1	*	39
e.	Sampler Flow Rate Measuring Device	1	*	39

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS (Continued)

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9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

- b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

#### 4.4.5.5 Reports

- a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.
- b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:
1. Number and extent of tubes inspected.
  2. Location and percent of wall-thickness penetration for each indication of an imperfection.
  3. Identification of tubes plugged.
- c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to 10 CFR 50.72(b)2(i) prior to resumption of plant operation. A report pursuant to 10 CFR 50.73(a)2(ii) shall be submitted to provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED  
DURING INSERVICE INSPECTION

Number of Steam Generators per Unit	Three
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One*

\*The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
	C-3	Perform action for C-3 result of first sample	N/A	N/A		
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G.  Prompt notification to NRC pursuant to 10 CFR 50.72 (b)2(i) and 10 CFR 50.73(a)2(ii)	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to 10 CFR 50.72 (b)2(i) and 10 CFR 50.73(a)2(ii)	N/A	N/A

$S = 3 \frac{N}{n} \%$  Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

## REACTOR COOLANT SYSTEM

### 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

#### LEAKAGE DETECTION SYSTEMS

#### LIMITING CONDITION FOR OPERATION

---

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

- a. A reactor building atmosphere particulate radioactivity monitoring system,
- b. The reactor building sump level, and
- c. Either the reactor building cooling unit condensate flow rate or a reactor building atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

#### ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

#### SURVEILLANCE REQUIREMENTS

---

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

- a. Reactor building atmosphere particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,
- b. Reactor building sump level-performance of CHANNEL CALIBRATION at least once per 18 months,
- c. Reactor building atmosphere gaseous radioactivity monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION, AND ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3
- d. Reactor building cooling unit condensate flow detector-performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

---

3.4.8 The specific activity of the primary coolant shall be limited to:

- a. Less than or equal to 1.0 microcurie per gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to  $100/\bar{E}$  microcuries per gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3\*:

- a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.
- b. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with  $T_{avg}$  less than 500°F within 6 hours.
- c. With the specific activity of the primary coolant greater than  $100/\bar{E}$  microcurie per gram, be in at least HOT STANDBY with  $T_{avg}$  less than 500°F within 6 hours.

\* With  $T_{avg}$  greater than or equal to 500°F.

## REACTOR COOLANT SYSTEM

### ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

- a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than  $100/\bar{E}$  microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission within 30 days. This report shall contain the results of the specific activity analyses together with the following information:
  1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,
  2. Fuel burnup by core region,
  3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,
  4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and
  5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

## SURVEILLANCE REQUIREMENTS

---

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

## RADIOACTIVE EFFLUENTS

### LIQUID HOLDUP TANKS

#### LIMITING CONDITION FOR OPERATION

---

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

- a. Condensate Storage Tank
- b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

#### ACTION:

- a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions or radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

## RADIOACTIVE EFFLUENTS

### SETTLING POND

#### LIMITING CONDITION FOR OPERATION

---

3.11.1.5 The quantity of radioactive material contained in each settling pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_j \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases, where,

$A_j$  = Pond inventory limit for single radionuclide "j", in curie.

$C_j$  = 10 CFR 20, Appendix B, Table II, column 2, concentration for single radionuclide "j", microcuries/ml.

$V$  = design volume of liquid and slurry in the pond, in gallons.

264 = Conversion unit, microcuries/curie per milliliter/gallon.

APPLICABILITY: At all times.

#### ACTION:

- a. With the quantity of radioactive material in the settling pond exceeding the above limit, immediately suspend all additions of radioactive material to the pond and within 48 hours reduce the pond contents to within the limit.
- b. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.11.1.5 The quantity of radioactive material contained in each batch of slurry (used powdex resin) to be transferred to the settling ponds shall be determined to be within the above limit by analyzing a representative sample of the slurry, and batches to be transferred to the settling ponds shall be limited by the expression:

$$\sum \frac{Q_j}{C_j} < 0.6$$

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

---

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

- a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.
- b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 1 hour and less than or equal to 2% by volume within 48 hours.
- c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

---

4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

## RADIOACTIVE EFFLUENTS

### GAS STORAGE TANKS

#### LIMITING CONDITION FOR OPERATION

---

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 160,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

#### ACTION:

- a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

## 3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

### 3/4.12.1 MONITORING PROGRAM

#### LIMITING CONDITION FOR OPERATION

---

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

- a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.
- b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \dots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

- c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.
- d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## RADIOLOGICAL ENVIRONMENTAL MONITORING

### SURVEILLANCE REQUIREMENTS

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4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

POWER DISTRIBUTION LIMITS

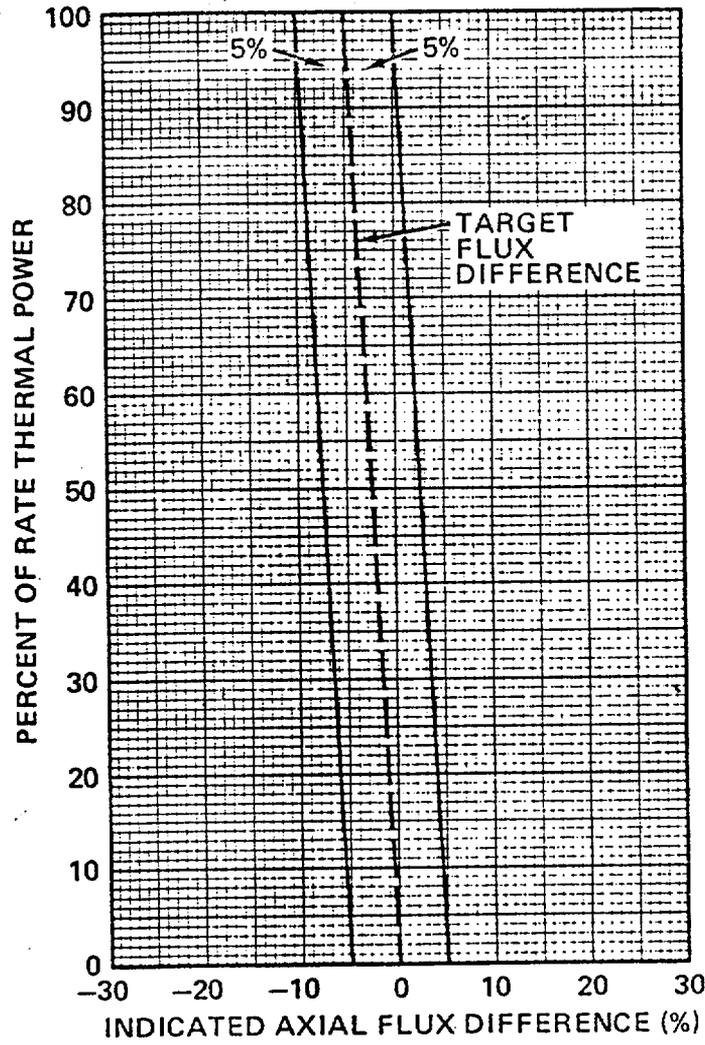


FIGURE B 3/4 2-1  
INDICATED AXIAL FLUX DIFFERENCE  
VERSUS THERMAL POWER

## POWER DISTRIBUTION LIMIT

### BASES

#### HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

- c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.
- d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}^N$  will be maintained within its limits provided conditions a. through d. above are maintained. As noted on Figures 3.2-3 and 3.2-4, RCS flow rate and  $F_{\Delta H}^N$  may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured  $F_{\Delta H}^N$  is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of  $F_{\Delta H}^N$  as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

$R_1$ , as calculated in 3.2.3 and used in Figure 3.2.3, accounts for  $F_{\Delta H}^N$  less than or equal to 1.49. This value is used in the various accident analyses where  $F_{\Delta H}^N$  influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.  $R_2$ , as defined, allows for the inclusion of a penalty for rod bow on DNBR only. Thus knowing this "as measured" values of  $F_{\Delta H}^N$  and RCS flow allows for "tradeoffs" in excess of  $R$  equal to 1.0 for the purpose of offsetting the rod bow DNBR penalty.

When an  $F_Q$  measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The radial peaking factor  $F_{xy}(Z)$  is measured periodically to provide assurance that the hot channel factor,  $F_0(Z)$ , remains within its limit. The  $F_{xy}$  limit for Rated Thermal Power ( $F_{xy}^{RTP}$ ) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.11 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS flow rate and  $F_{\Delta H}^N$  are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5% for RCS total flow rate and 4% for  $F_{\Delta H}^N$  have been allowed for in determination of the design DNBR value.

### 3/4.3 INSTRUMENTATION

#### BASES

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#### 3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1,  $Z + R + S < TA$ , the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip

## INSTRUMENTATION

### BASES

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#### REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite, or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident 1) safety injection pumps start and automatic valves position, 2) reactor trip, 3) feedwater isolation, 4) startup of the emergency diesel generators, 5) containment spray pumps start and automatic valves position, 6) containment isolation, 7) steam line isolation, 8) turbine trip, 9) auxiliary feedwater pumps start and automatic valves position, 10) containment cooling fans start and automatic valves position, 11) essential service water pumps start and automatic valves position, and 12) control room isolation and ventilation systems start.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR 50.72(b)2(i) prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

## REACTOR COOLANT SYSTEM

### BASES

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#### 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

##### 3/4.4.6.1 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," May 1973.

##### 3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 33 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

## ADMINISTRATIVE CONTROLS

- f. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.
- g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
- h. Review of all REPORTABLE EVENTS.
- i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.
- j. The plant Security Plan and changes thereto.
- k. The Emergency Plan and changes thereto.
- l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.
- m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Safety Review Committee.
- n. The unexpected offsite release of radioactive material and the report as described in 10 CFR 50.73.
- o. Changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

### AUTHORITY

6.5.1.7 The Plant Safety Review Committee shall:

- a. Recommend in writing to the Director, Nuclear Plant Operations approval or disapproval of items considered under 6.5.1.6a, c, d, e, j, and k above.
- b. Render determinations in writing to the Director, Nuclear Plant Operations with regard to whether or not each item considered under 6.5.1.6a, c, and d above constitutes an unreviewed safety question.
- c. Make recommendations in writing to the Director, Nuclear Plant Operations that actions reviewed under 6.5.1.6(b) above did not constitute an unreviewed safety question.
- d. Provide written notification within 24 hours to the Vice President, Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the Director, Nuclear Plant Operations however, the Director, Nuclear Plant Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

### RECORDS

6.5.1.8 The Plant Safety Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Vice President Nuclear Operations and the Chairman of the Nuclear Safety Review Committee.

## ADMINISTRATIVE CONTROLS

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### MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

### QUORUM

6.5.2.6 A quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 3 NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

### REVIEW

6.5.2.7 The NSRC shall review:

- a. The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.
- b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
- c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
- d. Proposed changes to Technical Specifications or this Operating License.
- e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
- f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.
- g. All REPORTABLE EVENTS.
- h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.
- i. Reports and meetings minutes of the Plant Safety Review Committee.

## ADMINISTRATIVE CONTROLS

- c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment.
- d. Events reportable pursuant to the Technical Specification 6.9 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the event and which provides recommendations to prevent recurrence. Such report shall be approved by the Director, Nuclear Plant Operations and forwarded to the Chairman of the Nuclear Safety Review Committee.
- e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c and 6.5.3.1.d shall be members of the plant staff that meet or exceed the qualification requirements of Section 4.4 of ANSI 18.1, 1971, as previously designated by the Director, Nuclear Plant Operations. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.
- f. Each review will include a determination of whether or not an unreviewed safety question is involved.

### RECORDS

6.5.3.2 Records of the above activities shall be provided to the Director, Nuclear Plant Operations, PSRC and/or NSRC as necessary for required reviews.

### 6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

- a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and
- b. Each REPORTABLE EVENT shall be reviewed by the PSRC and the results of this review shall be submitted to the NSRC and the Vice President, Nuclear Operations.

### 6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President, Nuclear Operations and the NSRC shall be notified within 24 hours.
- b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.
- c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC and the Vice President, Nuclear Operations within 14 days of the violation.

## ADMINISTRATIVE CONTROLS

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### 6.9 REPORTING REQUIREMENTS

#### ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator Office of Inspection and Enforcement unless otherwise noted.

#### STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

## ADMINISTRATIVE CONTROLS

6.9.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4) during the report. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. Historical annual average meteorology or meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

- a. Container volume,
- b. Total curie quantity (specify whether determined by measurement or estimate),
- c. Principal radionuclides (specify whether determined by measurement or estimate),
- d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

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- e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
- f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the Process Control Program (PCP) made during the reporting period.

### MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

### RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The  $F_{xy}$  limit for Rated Thermal Power ( $F_{xy}^{RTP}$ ) shall be provided to the Regional Administrator of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support  $F_{xy}^{RTP}$  will be by request from the NRC and need not be included in this report.

### SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

### 6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

## ADMINISTRATIVE CONTROLS

6.10.1 The following records shall be retained for at least five years:

- a. Records and logs of unit operation covering time interval at each power level.
- b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
- c. All REPORTABLE EVENTS submitted to the Commission.
- d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
- e. Records of changes made to the procedures required by Specification 6.8.1.
- f. Records of radioactive shipments.
- g. Records of sealed source and fission detector leak tests and results.
- h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

- a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.
- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Records of radiation exposure for all individuals entering radiation control areas.
- d. Records of gaseous and liquid radioactive material released to the environs.
- e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.
- f. Records of reactor tests and experiments.
- g. Records of training and qualification for current members of the unit staff.
- h. Records of in-service inspections performed pursuant to these Technical Specifications.

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- i. Records of Quality Assurance activities as specified in the NRC's approved SCE&G position on Regulatory Guide 1.88, Rev. 2, October 1976.
- j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- k. Records of meetings of the PSRC and the NSRC.
- l. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.
- m. Records of secondary water sampling and water quality.
- n. Records of analysis required by the radiological environmental monitoring program.

### 6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

### 6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).<sup>\*</sup> Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.
- c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

<sup>\*</sup>Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they otherwise comply with approved radiation protection procedures for entry into high radiation areas.

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6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area. The maximum allowable stay time for individuals in that area shall be established prior to entry. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem\*\* that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP direct or remote (such as use of closed circuit TV cameras) continuous surveillance shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

### 6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:
  - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;
  - b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
  - c. Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.
2. Shall become effective upon review and acceptance as set forth in 6.5 above.

### 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

\*\*Measurement made at 18" from source of radioactivity.

## ADMINISTRATIVE CONTROLS

### 6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:
  - a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);
  - b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and
  - c. Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.
2. Shall become effective upon review and acceptance as set forth in 6.5 above.

### 6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

#### 6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Plant Safety Review Committee. The discussion of each change shall contain:
  - a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;
  - b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
  - c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;
  - d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differs from those previously predicted in the license application and amendments thereto;

## ADMINISTRATIVE CONTROLS

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- e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;
  - f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;
  - g. An estimate of the exposure to plant operating personnel as a result of the change; and
  - h. Documentation of the fact that the change was reviewed and found acceptable by the PSRC.
2. Shall become effective upon review and acceptance as set forth in 6.5 above.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NO.35 TO FACILITY OPERATING LICENSE NPF-12  
SOUTH CAROLINA ELECTRIC & GAS COMPANY  
SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

I. INTRODUCTION

By letter dated February 22, 1984, South Carolina Electric and Gas Company requested a change to Technical Specification reporting requirements to be in accordance with new regulation 10 CFR 50.73.

II. EVALUATION

The new 10 CFR 50.73 provides for a revised Licensee Event Report System and replaces all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications. We have reviewed the requested change and find that it correctly reflects 10 CFR 50.73. Therefore, we conclude that the requested change is acceptable.

III. ENVIRONMENTAL CONSIDERATION

This amendment involves a change in recordkeeping and reporting requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

IV. CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (49 FR 38408) on September 28, 1984, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have concluded, based on the considerations discussed above, that:  
(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and  
(2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Jon B. Hopkins, Licensing Branch No. 4, DL

Dated: January 2, 1985

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P PDR

January 2, 1985

AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. NPF-12 - Virgil C. Summer Unit 1

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