March 15, 1990

Docket No. 50-395

DISTRIBUTION

See attached sheet

Mr. O. S. Bradham
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, South Carolina 29065

Dear Mr. Bradham:

SUBJECT: ISSUANCE OF AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-12 - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1, REGARDING REMOVAL OF THE RESISTANCE TEMPERATURE DETECTOR (RTD) BYPASS MANIFOLD SYSTEM HOT AND COLD LEG PIPING (TAC NO. 74826)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 90 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 21, 1989, as supplemented in letters dated December 11, 1989, January 2, 1990, and February 6, 1990.

Your July 21, 1989 submittal requested a revision to TS. 2.2.1, Reactor Trip System Instrumentation Setpoints, TS 3/4.3.1, Reactor Trip System Instrumentation and TS 3/4.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation. This revision changes certain setpoints and allowable values in the above TS as a result of the replacement of the resistance temperature device (RTD) bypass manifold with fast response RTDs located in reactor coolant hot leg and cold leg piping.

This Amendment approves this TS change. A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's Bi-weekly <u>Federal</u> <u>Register</u> notice.

Sincerely,
Original Signed By:

John J. Hayes, Jr., Project Manager
Project Directorate II-1
Division of Reactor Projects I/II



Enclosures:
1. Amendment No. 90 to NPF-12
2. Safety Evaluation

cc w/enclosures
See next page

DFo1 1/1

| OFC | :LA:PD21-1 | :PM:PD21-1 | :D:PD21-1 | : | : | : |
|---|---|---|---|---|---|---|
| NAME | :PAnderson | :JHayes:dt | :EAdensam | : | : | : |
| DATE | :3/5/90 | :3/5/90 | :3/15/90 | : | : | : |

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Document Name: SUMMER AMENDMENT 74826

Mr. O. S. Bradham
South Carolina Electric & Gas Company

Virgil C. Summer Nuclear Station

cc:

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AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT 1

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
J. Hayes
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
J. Partlow (9A2)
T. Barnhart (4) (P1-137)
W. Jones (P-130A)
E. Butcher (11F23)
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

DF01
1/1

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. 90
License No. NPF-12

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

   A. The application for amendment by South Carolina Electric & Gas Company (the licensees), dated July 21, 1989, as supplemented December 11, 1989 January 2, 1990 and February 6, 1990, 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 changes to the Technical Specifications, as indicated in the attachment 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 and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 90 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance, and shall be implemented on restart following the March 23, 1990 refueling outage and prior to entering the modes specified in Tables 3.3-1 and 3.3-3 as applicable for the changes in this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Lester Kintner/for

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 15, 1990

*SEE PREVIOUS CONCURRENCE

| OFC | LA:PD21:DRPR | PM:PD21:DRPR | OGC * | D:PD21:DRPR | | | |
|---|---|---|---|---|---|---|---|
| NAME | PAnderson | JHayes:sw | BMBordenick | EAdensam | | | |
| DATE | 3/15/90 | 3/12/90 | 3/12/90 | 3/15/90 | | | |

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ATTACHMENT TO LICENSE AMENDMENT NO. 90

TO 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.

| Remove Pages | Insert Pages |
|---|---|
| 2-5 | 2-5 |
| 2-8 | 2-8 |
| 2-9 | 2-9 |
| 2-10 | 2-10 |
| B 2-5 | B 2-5 |
| 3/4 3-9 | 3/4 3-9 |
| 3/4 3-27 | 3/4 3-27 |
| 3/4 3-28b | 3/4 3-28b |

## TABLE 2.2-1

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| Functional Unit | Total Allowance (TA) | Z | S | Trip Setpoint | Allowable Value |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | Not Applicable | NA | NA | NA | NA |
| 2. Power Range, Neutron Flux High Setpoint | 7.5 | 4.56 | 0 | $\leq$109% of RTP | $\leq$111.2% of RTP |
| Low Setpoint | 8.3 | 4.56 | 0 | $\leq$25% of RTP | $\leq$27.2% of RTP |
| 3. Power Range, Neutron Flux High Positive Rate | 1.6 | 0.5 | 0 | $\leq$5% of RTP with a time constant $\geq$2 seconds | $\leq$6.3% of RTP with a time constant $\geq$2 seconds |
| 4. Power Range, Neutron Flux High Negative Rate | 1.6 | 0.5 | 0 | $\leq$5% of RTP with a time constant $\geq$2 seconds | $\leq$6.3% of RTP with a time constant $\geq$2 seconds |
| 5. Intermediate Range, Neutron Flux | 17.0 | 8.4 | 0 | $\leq$25% of RTP | $\leq$31% of RTP |
| 6. Source Range, Neutron Flux | 17.0 | 10.0 | 0 | $\leq 10^5$ cps | $\leq 1.4 \times 10^5$ cps |
| 7. Overtemperature ΔT | 9.8 | 7.21 | 1.6 & 1.2** | See note 1 | See note 2 |
| 8. Overpower ΔT | 5.2 | 1.96 | 1.6 | See note 3 | See note 4 |
| 9. Pressurizer Pressure-Low | 3.1 | 0.71 | 1.5 | $\geq$1870 psig | $\geq$1859 psig |
| 10. Pressurizer Pressure-High | 3.1 | 0.71 | 1.5 | $\leq$2380 psig | $\leq$2391 psig |
| 11. Pressurizer Water Level-High | 5.0 | 2.18 | 1.5 | $\leq$92% of instrument span | $\leq$93.8% of instrument span |
| 12. Loss of Flow | 2.5 | 1.48 | .6 | $\geq$90% of loop design flow* | $\geq$88.9% of loop design flow* |

*Loop design flow = 96,500 gpm
RTP - RATED THERMAL POWER
**1.6% span for Delta-T (RTDs) and 1.2% for Pressurizer Pressure.

## TABLE 2.2-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

## NOTATION

NOTE 1: OVERTEMPERATURE $\Delta T$

$$\Delta T \leq \Delta T_o \left[K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} [T - T'] + K_3(P - P') - f_1(\Delta I)\right]$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured $\Delta T$ by RTD Instrumentation |
| $\Delta T_o$ | $\leq$ | Indicated $\Delta T$ at RATED THERMAL POWER |
| $K_1$ | $\leq$ | 1.203 |
| $K_2$ | $\geq$ | 0.03006 |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | The function generated by the lead-lag controller for $T_{avg}$ dynamic compensation |
| $\tau_1$, $\tau_2$ | = | Time constants utilized in lead-lag controller for $T_{avg}$, $\tau_1 \geq 28$ secs., $\tau_2 \leq 4$ secs. |
| T | = | Average temperature, °F |
| T' | $\leq$ | 587.4°F Reference $T_{avg}$ at RATED THERMAL POWER |
| $K_3$ | $\geq$ | 0.00147 |
| P | = | Pressurizer pressure, psig |
| P' | $\geq$ | 2235 psig, Nominal RCS operating pressure |
| S | = | Laplace transform operator, $sec^{-1}$. |

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

and $f_1$ ($\Delta I$) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 24 percent and + 4 percent $f_1$ ($\Delta I$) = 0 where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -24 percent, the $\Delta T$ trip setpoint shall be automatically reduced by 2.27 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +4 percent, the $\Delta T$ trip setpoint shall be automatically reduced by 2.13 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.2 percent $\Delta T$ Span.

NOTE 3: OVERPOWER $\Delta T$

$$\Delta T \leq \Delta T_o \, [K_4 - K_5 \, \left(\frac{\tau_1 S}{1 + \tau_3 S}\right) T - K_6 \, [T - T'']]$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | as defined in Note 1 |
| $\Delta T_o$ | = | as defined in Note 1 |
| $K_4$ | $\leq$ | 1.0875 |
| $K_5$ | $\geq$ | 0.02/°F for increasing average temperature and 0 for decreasing average temperature |
| $\frac{\tau_3 S}{1 + \tau_3 S}$ | = | The function generated by the rate-lag controller for $T_{avg}$ dynamic compensation |

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 3 (continued)

| | | |
|---|---|---|
| $T_3$ | = | Time constant utilized in the rate-lag controller for $T_{avg}$, $\tau_3 \geq 10$ secs. |
| $K_6$ | $\geq$ | $0.00156/°F$ for $T > T''$ and $K_6 = 0$ for $T \leq T''$ |
| T | = | as defined in Note 1 |
| T" | $\leq$ | 587.4°F Reference $T_{avg}$ at RATED THERMAL POWER |
| S | = | as defined in Note 1 |

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.4 percent ΔT Span.

Intermediate and Source Range, Nuclear Flux (Continued)

uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a reactor trip at about $10^{+5}$ counts per second unless manually blocked when P-6 becomes active. The purpose of the P-6 setpoint, which is above the lower end of the intermediate range scale, is to give the operators sufficient time to actuate the source range reactor trip block. The Intermediate Range channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds) plus thermal delays associated with the RTD's mounted in the thermowells (about 5 seconds), and pressure is within the range between the Pressurizer high and low pressure trips. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, 2) pressurizer pressure, and 3) axial power distribution. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower delta T trip provides assurance of fuel integrity (e.g., no fuel melting and less than 1 percent cladding strain) under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and 2) rate of change of temperature for dynamic compensation for piping and thermal delays from the core to the loop temperature detectors to ensure that the allowable heat generation rate (Kw/ft) is not exceeded. The overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP 9226, "Reactor Core Response to Excessive Secondary Steam Break."

Pressurizer Pressure

In each of the pressure channels, there are two independent bistables, each with its own trip setting to provide for a high and low pressure trip thus limiting the pressure range in which reactor operation is permitted. The low setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

# TABLE 3.3-2

## REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
| --- | --- |
| 1. Manual Reactor Trip | Not Applicable |
| 2. Power Range, Neutron Flux | ≤ 0.5 seconds(1) |
| 3. Power Range, Neutron Flux, High Positive Rate | Not Applicable |
| 4. Power Range, Neutron Flux, High Negative Rate | ≤ 0.5 seconds(1) |
| 5. Intermediate Range, Neutron Flux | Not Applicable |
| 6. Source Range, Neutron Flux | Not Applicable |
| 7. Overtemperature ΔT | ≤ 8.5 seconds(1)(2) |
| 8. Overpower ΔT | ≤ 8.5 seconds(1)(2) |
| 9. Pressurizer Pressure--Low | ≤ 2.0 seconds |
| 10. Pressurizer Pressure--High | ≤ 2.0 seconds |
| 11. Pressurizer Water Level--High | Not Applicable |

(1)Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.
(2)The 8.5 second response time includes a 5.0 second delay for the RTDs mounted in thermowells.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| Functional Unit | Total Allowance (TA) | Z | S | Trip Setpoint | Allowable Value |
|---|---|---|---|---|---|
| 4. STEAM LINE ISOLATION | | | | | |
| a. Manual | NA | NA | NA | NA | NA |
| b. Automatic Actuation Logic and Actuation Relays | NA | NA | NA | NA | NA |
| c. Reactor Building Pressure-High 2 | 3.0 | 0.71 | 1.5 | ≤6.35 | ≤6.61 |
| d. Steam Flow in Two Steamlines-High, Coincident with | 20.0 | 13.16 | 1.5/ 1.5 | ≤ a function defined as follows: A ΔP corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load | ≤ a function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 114.0% of full steam flow at full load. |
| $T_{avg}$ - Low-Low | 4.0 | .71 | .8 | ≥552.0°F | ≥548.4°F |
| e. Steamline Pressure - Low | 20.0 | 10.71 | 1.5 | ≥675 psig | ≥635 psig(1) |

(1) Time constants utilized in lead lag controller for steamline pressure low are as follows:
$\tau_1 \geq 50$ secs. $\tau_2 \leq 5$ secs.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| Functional Unit | Total Allowance (TA) | Z | S | Trip Setpoint | Allowable Value |
|---|---|---|---|---|---|
| 9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS | | | | | |
| INTERLOCKS | | | | | |
| a. Pressurizer Pressure, P-11 | 3.1 | .71 | 1.5 | 1985 psig | ≥1974 psig & ≤1996 psig |
| b. $T_{avg}$ Low-Low, P-12 | 4.0 | .71 | .8 | 552°F | ≥548.4°F & ≤555.6°F |
| c. Reactor Trip, P-4 | NA | NA | NA | NA | NA |

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

## 1.0 INTRODUCTION

By letter dated July 21, 1989, South Carolina Electric & Gas Company (the licensee) proposed to modify the reactor coolant temperature measurement system for the hot and cold legs for V.C. Summer Nuclear Station Unit No. 1, (Summer) and requested changes to the Summer Technical Specifications (TS).

The proposed modification would eliminate the resistance temperature device (RTD) bypass manifold and replace it with fast response RTDs located in reactor coolant hot leg and cold leg piping. This modification would eliminate operating obstacles associated with the bypass system. These obstacles include leakage through flanges and valves and radiation exposure during reactor building maintenance. In addition, the licensee proposed changes to TS 2.2.1, Reactor Trip System Instrumentation Setpoints, TS 3/4.3.1, Reactor Trip System Instrumentation, and TS 3/4.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation which would revise certain setpoints, allowable values, and response times as a result of the replacement of the RTD bypass manifold.

## 2.0 BACKGROUND

The July 21, 1989 submittal contained the proposed TS changes and the Westinghouse Proprietary (WCAP-12189) and Non-Proprietary (WCAP-12190), reports covering the technical basis for the modification to the reactor coolant temperature measurement system for the hot and cold legs. These are not topical reports but are plant specific. The December 11, 1989 submittal contained the plant specific safety evaluation for the change. Clarifying information in support of the amendment request was submitted on January 2, 1990 and February 6, 1990.

The current method of measuring the hot and cold leg reactor coolant temperature uses an RTD bypass system. This system was designed to address temperature streaming in the hot legs and, by use of shutoff valves, to allow replacement of the direct immersion narrow-range RTDs without draindown of the Reactor Coolant System (RCS). For increased accuracy in measuring the hot leg temperatures, sampling scoops were placed in each hot leg at three locations of a cross section, 120 degrees apart. The flow from the scoops is piped to a manifold where a direct immersion RTD measures the average temperature of the flow from the three scoops. This bypass flow is routed back to the RCS downstream of the steam generator. The cold leg temperature is measured in a similar manner with piping to a bypass manifold except that no scoops are used, as temperature streaming is not a problem due to the mixing action of the RCS pump. Both hot and cold leg manifolds empty through a common header to the intermediate leg between the steam generator and reactor coolant pump.

The output from the bypass loop RTDs provides the signals necessary to calculate $T_{avg}$ and Delta T. The $T_{avg}$ and Delta T parameters are then input to the reactor protection system overtemperature and overpower Delta T reactor trips. The control system input of $T_{avg}$ and Delta T are provided through a separate dedicated set of bypass loop RTDs and $T_{avg}$ and Delta T calculations.

The new method proposed for measuring the hot and cold leg temperature uses narrow-range fast response RTDs manufactured by Weed Instruments, Inc. The RTDs are placed in thermowells to allow replacement without draindown. The thermowells, however, increase the response time.

The licensee proposed to measure the hot leg temperature on each loop with three fast response, narrow range, dual element RTDs mounted in thermowells. One element of the RTD is considered active while the other element is held in reserve as a spare. To accomplish the sampling function of the RTD bypass manifold system and the need for additional hot leg piping penetrations, the thermowells will be placed within the existing scoops. A hole will be drilled through the end of each scoop so that water will flow through the existing holes in the leading edge of the scoop, past the RTD, and out through the hole. This RTD arrangement accomplishes the same sampling/temperature averaging function as the bypass manifold system.

The RTD measures the temperature at one point in the new method. This is in contrast to the RTD bypass flow method which utilizes the temperature measurement of the average of the flow from the five sample holes from the hot leg scoops. In the new method, the radial location of each RTD measurement is at the same radius as the center hole of the scoop. Therefore, it is the equivalent of the average scoop sample if a linear radial temperature gradient exists in the pipe.

The licensee has also proposed to modify the means for measuring the cold leg temperature. Because temperature gradients in the cold leg are eliminated by the mixing action of the reactor coolant pumps, only one RTD is necessary for cold leg temperature measurement. The licensee proposes to mount a single thermowell with one fast response, narrow range, duel-element RTD in each cold leg at the discharge of the reactor coolant pump. One of the dual elements is a spare. This is in place of the original method in which the measurement was by an external RTD in the cold leg bypass manifold. As in the hot leg, the bypass manifold penetration nozzle will be modified to accept the RTD thermowell. Additionally, the bypass manifold return line will be capped at the nozzle on the intermediate leg.

The proposed modification would revise the reactor protection system $T_{avg}$ and Delta T input methodology and the corresponding response times and instrument uncertainties. System input of $T_{avg}$ and Delta T to plant control systems will also be modified. A microprocessor-based system is used to perform the averaging of the reactor coolant hot leg signals from the three RTDs in each hot leg and then to transmit the signal for the average hot leg temperature to protection and control systems. There is a routine for performing a quality check of the three temperature signals for the loss of one of the three RTD sensor inputs. The bias considers the past history of the previous hot leg readings.

Neither of these modifications will affect the single existing wide range RTDs installed in each hot and cold leg of the reactor coolant system. These RTDs will continue to provide hot and cold leg temperature information for reactor start-up, shutdown or post-accident monitoring.

The staff has reviewed the application and supplementary material and prepared the following evaluation.

## 3.0 EVALUATION

The evaluation which was performed for the RTD bypass manifold replacement covered three major areas. These were conformance with the ASME Code, instrumentation and control requirements and meeting accident analysis requirements. The following sections present the evaluations associated with each of these areas.

### 3.1 Conformance With ASME Code

The replacement of the presently installed RTD bypass system with fast acting narrow range RTD thermowells requires modifications to the hot leg scoops, the hot leg piping, the crossover leg bypass return nozzle, and the cold leg bypass manifold connection. The licensee has stated that all welding and NDE will be performed in accordance with ASME Code, Section XI, requirements.

The licensee has proposed to remove the original three scoops in the hot legs of reactor coolant loops A, B and C which feed the bypass manifold and the bypass manifold connection and to modify all scoops to accept three fast response RTD thermowells and to provide the proper flow path. The licensee has proposed that a thermowell design will be used and that the thermowell will be fabricated in accordance with Section III (Class 1) of the ASME Code. The installation of the thermowell into the scoop will be performed using gas tungsten arc weld (GTAW) for the root pass and finished out with either GTAW or shielded metal arc weld (SMAW). The welding will be examined using a penetrant test (PT) in accordance with the ASME Code, Section XI. Prior to welding, the surface of the scoop onto which welding will be performed will be examined as required by Section XI of the ASME Code. The cold leg RTD bypass line will be removed. The nozzle will then be modified to accept the fast response RTD thermowell. The installation of the thermowell into the nozzle will be performed using GTAW for the root pass and finished with either GTAW or SMAW. Weld inspection by PT will be performed as required by Section XI.

the thermowells will extend into the flow stream to a depth that has been justified based on analysis. The root weld joining the thermowells to the modified nozzle will be deposited with GTAW and the remainder of the weld may be deposited with GTAW or SMAW. Penetrant testing will be performed in accordance with Section XI. The thermowells will be fabricated in accordance with the ASME Code, Section III (Class 1).

The cross-over leg bypass return piping connection will be removed and the nozzles capped. The cap design, including materials, will meet the pressure boundary criteria of ASME Section III (Class 1). The cap will be root welded to the nozzles by GTAW and fill welded by either GTAW or SMAW. Non-destructive examinations, i.e., PT and radiographs, will be performed in accordance with ASME Section XI. Machining of the bypass return nozzle, as well as any machining performed during modification of the penetrations in the hot and cold legs, shall be performed such as to minimize debris escaping into the reactor coolant system.

In accordance with Article IWA-4000 of Section XI of the ASME Code, a hydrostatic test of new pressure boundary welds is required when the connection to the pressure boundary is larger than one inch in diameter. Because the cap for the crossover leg bypass return pipe and the cold leg RTD connections are larger than one inch in diameter, a system hydrostatic test is required after the bypass elimination modification is complete. Paragraph IWB-5222 of Section XI defines this test pressure to be 1.02 times the normal operating pressure at a temperature of 500°F or greater. The hydrotest temperature will be achieved by pump heat.

The staff has reviewed the information supplied by the licensee and has determined that the integrity of the reactor coolant piping as a pressure boundary component will be maintained by adhering to the applicable ASME Code sections and General Design Criteria. Further, the pressure retaining capability and fracture prevention characteristics of the piping will not be compromised by these modifications. Therefore, the staff concludes that the modifications will be done in accordance with the applicable code and regulatory requirements and is acceptable.

### 3.2 Accident Analyses

The increased response time for the proposed RTD design impacts the results of the accident analyses. The hot leg temperature measurement could potentially affect the accident analyses and is the principal contributor in the analyses for calculating the RCS flow measurement uncertainty.

#### 3.2.1 RTD Response Time

The overall response time requirement for the new thermowell RTD temperature system is two and one-half seconds longer than the former RTD bypass system (8.5 vs 6.0 seconds). The 8.5 second overall response time for the new RTD system is a result of adding an electronics delay to the response time for the RTD sensor and including a margin for uncertainties. Because of the increased channel response time, there are longer delays from the time when fluid conditions in the RCS require overtemperature delta-T (OTDT) or overpower delta-T (OPDT) reactor trips until a trip is actually generated. The licensee presented information in WCAP-12189 concerning the Final Safety Analysis Report (FSAR), Chapter 15, non-LOCA accidents that rely on the above mentioned trips. These accidents were evaluated for the longer response time. This longer response time was approved in the staff's Safety Evaluation for the transition reload core which was issued with Amendment No. 75 to the Summer Operating License.

As noted in NUREG-0809, "Safety Evaluation Report, Review of Resistance Temperature Detector Time Response Characteristics," extensive RTD testing has revealed RTD time response degradation with aging. In view of this, surveillance tests are needed. The approved in-situ method for measuring RTD response time is the loop current step response (LCRS), method. By letter dated January 2, 1990, the licensee confirmed that LCRS tests will be performed prior to initial criticality and the RTDs will be checked on a refueling basis not to exceed 18 months. This is acceptable to the staff.

The staff determined, in a previous amendment, that the increased response time was acceptable. The staff has also determined, with respect to this amendment request, that the planned RTD response time test and test frequency to identify RTD time response degradation, is acceptable.

3.2.2. RTD Uncertainty

The new method of measuring each hot leg temperature with three thermowell RTDs in place of the RTD bypass system with three scoops has been analyzed to be slightly more accurate. The new RTD thermowell with measurement at one point may have a small streaming error relative to the former scoop flow measurement because of a temperature gradient over the 5-inch scoop span. However, this gradient has been calculated to have a small effect. Also, since possible temperature uncertainties from imbalanced scoop flows are eliminated, the overall result is more effective. Since the new method uses three RTDs for each hot leg temperature measurement, it is statistically a more accurate temperature measurement than the former method which used only one RTD for each new hot leg temperature measurement. System uncertainty calculations were performed by the licensee to establish appropriate changes to be made in the reactor protection system setpoints to account for the revised uncertainties. The setpoint changes are identified in the proposed changes made to Tables 2.2-1 and 3.3-4 of the Summer TS.

In the January 2, 1990 submittal, the licensee has stated that the hot leg temperature measured with the thermowell RTDs can be compared with the hot leg temperature previously measured with the RTDs in the bypass system by comparing Delta-T measurements normalized to full power. This comparison can be utilized to verify that the new method of temperature measurement is accurate. This is similar to the approach taken at other facilities for a similar modification and is acceptable to the staff.

The staff has determined that the uncertainties associated with the temperature measurements made by the thermowell RTDs are acceptable.

### 3.2.3 Non-LOCA Accidents

The primary impact of the RTD bypass system elimination is the increased RTD response time. Thus, only those events which rely on the OTDT and OPDT reactor trips are potentially impacted. The accidents in the FSAR, Sections 15.1 to 15.6, were examined by the licensee and it was determined that plant operation will be maintained within the bounds of safe, analyzed conditions as established by the reanalysis performed for the current transition core reload associated with Amendment No. 75. Since the increased RTD response time of 8.5 seconds was used in the prior reanalysis, the staff finds the previously reviewed results remain acceptable.

In summary, the staff has evaluated the impact of the RTD bypass elimination for V.C. Summer on the FSAR non-LOCA accident analyses has been evaluated. For the events impacted, it was demonstrated that the conclusions presented in the transition core reload, Amendment No. 75, remain valid.

### 3.2.4 LOCA Evaluation

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The information presented by the licensee in WCAP-12189 that the magnitude of the uncertainties are such that RCS inlet and outlet temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will not be affected. Past sensitivity studies concluded that the inlet temperature effect on peak clad temperature is dependent on break size. As a result of these studies, the LOCA analyses are performed at a nominal value of inlet temperature without consideration of small uncertainties. The RCS flow rate and steam generator secondary side temperature and pressure are also determined using the loop average temperature ($T_{avg}$) output. These nominal values used as inputs to the analyses are not affected due to the RTD bypass elimination.

The staff has concluded that the elimination of the RTD bypass piping will not affect the LOCA analyses input and, hence, the results of the analyses remain unaffected. Therefore, the staff finds the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint.

### 3.2.5 Flow Measurement Uncertainty

The licensee stated in their January 2, 1990 submittal that the flow calculation uncertainty and cold leg elbow tap flow uncertainty values are based on a prior evaluation for Summer which was submitted in a November 21, 1986 letter to the NRC. This evaluation was used to establish the present TS value of 2.1 percent for RCS loop flow uncertainty. When compared to the value of 1.9 percent calculated in WCAP-12189, the present value of 2.1 percent is judged to be sufficiently conservative to accommodate uncertainties associated with feed flow venturi fouling (typically 0.1 percent). Also, the licensee has previously committed to cleaning the feed flow venturies and measuring the throat diameter every refueling outage.

Based upon the above, the staff has determined that the licensee's approach to retaining the value of 2.1 percent for the RCS loop flow uncertainty is acceptable.

### 3.2.6 Summary, Accident Analyses

The impact of the RTD bypass elimination for Summer on the FSAR, Chapter 15, non-LOCA accident analyses has been evaluated. For the events impacted by the increase in the channel response time, it has been demonstrated that the conclusions presented in the evaluation of the transition reload core associated with Amendment No. 75 of the Summer license remain valid. For the remaining Chapter 15 non-LOCA events, the effect of the increased initial RCS average temperature error allowance has been ascertained by separate evaluations. In all instances, the conclusions presented in the Summer FSAR remain valid under this error allowance assumption and the DNBR limit value is met. The current LOCA analysis is based on conservative nominal input values and remains acceptable. The licensee's analysis to support an RCS flow measurement uncertainty value, which includes the new hot leg RTD temperature accuracy, was provided in WCAP-12189, and is acceptable.

### 3.3 Instrumentation and Control

To accomplish the hot leg temperature averaging function previously done by the bypass manifold system, the modified hot leg RTD temperature signals are electronically averaged in the reactor protection system. This averaged T-hot signal will then be used with the T-cold signal to calculate reactor coolant loop Delta T and $T_{avg}$ values utilized in the reactor protection and control systems.

The present bypass system uses separate dedicated RTDs for the control and protection systems. However, the modified system thermowell mounted RTDs will be used for both protection and control. This class IE to Non-IE interface will require the use of isolation devices for control system $T_{avg}$ and Delta T signals derived from the reactor protection system. $T_{avg}$ and Delta T signals used in the control grade logic will be input into an analog median signal selector (MSS). This device selects the signal between the highest and lowest values of the three $T_{avg}$ and Delta T loop inputs. By selecting the median value, the MSS provides the plant control system with a valid $T_{avg}$ and Delta T in the event that a spurious signal (failed sensor or drift error) is input to the MSS. The MSS also preserves the functional independence between separate control and protection systems that share a common sensor.

To ensure proper operation of the MSS, the existing manual switches that allow for defeating a $T_{avg}$ or Delta T signal from a single loop will be eliminated. Also, the conversion to thermowell mounted RTDs will result in the elimination of the control grade RTDs and their associated control board indicators. The protection system channels will now provide inputs to the control system through isolators and the analog MSS. Existing control board alarms, indicators, $T_{avg}$ and Delta T deviation alarms will provide the means to identify RTD failures. A failure of a hot leg RTD may be handled in two ways. The first puts the affected channel in trip and rescales the electronics to average the remaining two inputs. A bias is then added to the T-hot average signal in order to compensate for the failed RTD and to maintain a value comparable with the previous three RTD average. As an alternate, the failed RTD element may be disconnected and the spare element utilized. A cold leg RTD failure can be handled by utilizing the spare RTD element provided in each loop.

The staff has determined that the modified RTD system is not functionally different from the unmodified system except for the use of 3 RTDs instead of one. The reactor trip or engineered safety features actuation systems will operate as before. The original staff evaluation as documented in Section 7 of the Safety Evaluation for Summer remains valid. The additional electronics for averaging the three T-hot RTD signals (7300 based) will be qualified to the same level as the existing 7300 electronics. The isolation devices are also standard 7300 series equipment and were previously reviewed under Westinghouse Report WCAP 8892A. The RTD qualification will be to 10 CFR 50.49.

To support the modifications required to eliminate the RTD bypass manifold system, changes to the Summer TS were proposed. One of these changes is the result of the difference in response time of thermowell mounted RTDs as opposed to the original RTD bypass system. The staff has determined, based upon a review of calculations performed by the licensee and a review of the Summer TS response times for OTDT and OPDT, that the proposed changes to response times are acceptable.

The remaining technical specification revisions are a result of the difference in uncertainty considerations between the thermowell mounted RTDs and the bypass manifold system, i.e., calorimetric flow, hot leg temperature streaming and instrument uncertainties. Revised values for allowable value, Z and S for OTDT, OPDT, loss of flow, $T_{avg}$ low-low along with new set point values for P12 were calculated using essentially the same setpoint methodology as previously approved by the staff. Again, evaluations by the licensee determined that the revised values of allowable value, Z and S remain valid for the proposed bypass elimination. Based upon the staff's review of the licensee's submittals and the Summer TS, the staff finds the proposed plant modification to replace the RTD bypass manifold system with thermowell mounted, fast response RTDs mounted directly in the reactor coolant system to be acceptable.

## 4.0 TECHNICAL SPECIFICATION CHANGES

The licensee has proposed changes to Table 2.2-1 of TS 2.2.1, Reactor Trip System Instrumentation Setpoints, and Table 3.3-4 of TS 3/4.3.2, Engineered Safety Feature Actuation System Instrumentation to allow operation of Summer within acceptable safety limits. The changes are associated with trip setpoints for OTDT, OPDT, loss of flow, steam line isolation on low-low $T_{avg}$, and engineered safety feature interlocks on low-low $T_{avg}$. In addition, a footnote is added to Table 3.3-2, Reactor Trip System Instrumentation Response Times of TS 3/4.3.1, Reactor Trip System Instrumentation and a change is made to the Bases Section 2.2.1, Reactor Trip System Instrumentation Setpoints. These changes are necessary for consistency with the revised uncertainty analysis.

The staff finds the proposed change in values for the reactor trip setpoints along with the other proposed changes, based upon the above performed evaluations, acceptable.

## 5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the Surveillance Requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of this amendment.

## 6.0 CONCLUSION

The Commission has issued a "Notice of Consideration of Issuance of Amendment to Facility Operating License and Propose No Significant Hazards Consideration Determination and Opportunity for Hearing" which was published in the FEDERAL REGISTER on October 18, 1989 (54 FR 42864) and consulted with the State of South Carolina. No public comments or request for hearing were received, and the State of South Carolina did not have comments. The additional information provided by the licensee on December 11, 1989, January 2, 1990, and February 6, 1990 was clarifying information and did not change the substance of the Amendment request.

The staff has concluded, based upon 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.

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