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

August 29, 2000

Mr. Stephen A. Byrne
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATIONS CHANGES RELATED TO RESPONSE TIME TESTING ELIMINATION (TAC NO. MA8632)

Dear Mr. Byrne:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment changes the Technical Specifications in response to your application dated April 6, 2000. This amendment will eliminate the response time testing of the Reactor Trip System and the Engineered Safety Feature Actuation System.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Bi-weekly Federal Register notice. This completes the staff's efforts on TAC No. MA8632.

Sincerely,

Karen Cotton

Karen Cotton, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 146 to NPF-12
2. Safety Evaluation

cc w/enclosures: See next page

AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT NO. 1

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

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. 146
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 licensee), dated April 6, 2000, 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. 146 , 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 prior to the commencement of Refueling Outage 12, scheduled for October 4, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 29, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 146

TO FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1-3	1-3
1-5	1-5
3/4 3-1	3/4 3-1
3/4 3-15	3/4 3-15
B3/4 3-1a	B3/4 3-1a
B3/4 3-1b	B3/4 3-1b
B3/4 3-1c	B3/4 3-1c

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

PURGE - PURGING

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

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 2900 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. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

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.

1.30 Not Used

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.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor trip system instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the reactor trip system instrumentation surveillance requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be verified to be within its limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint Column but more conservative than the value shown in the Allowable Value Column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the engineered safety feature actuation system instrumentation surveillance requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within the limit at least once per 18 months. Each verification shall include at least one train such that both trains are verified at least once per 36 months and one channel per function such that all channels are verified at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

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

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. Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements, or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP 14036-P-A, Revision 1, "Elimination of Periodic Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component into operational service and re-verified following maintenance or modification that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for the repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing element of a transmitter.

The Engineered Safety Features response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 2 and 3) are based on values assumed in the non-LOCA safety analyses. These analyses are for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction isolation valves are closed following opening of the RWST charging pumps suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 1) the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

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

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.

Several automatic logic functions included in this specification are not necessary for Engineered Safety Feature System actuation but their functional capability at the specified setpoints enhances the overall reliability of the Engineered Safety Features functions. These automatic actuation Systems are purge and exhaust isolation from high containment radioactivity, turbine trip and feedwater isolation from steam generator high-high water level, initiation of emergency feedwater on a trip of the main feedwater pumps, automatic transfer of the suctions of the emergency feedwater pumps to service water on low suction pressure, and automatic opening of the containment recirculation sump suction valves for the RHR and spray pumps on low-low refueling water storage tank level.

The service water response time includes: 1) the start of the service water pumps and, 2) the service water pumps discharge valves (3116A,B,C-SW) stroking to the fully opened position. This condition of the valves assures that flow will become established through the component cooling water heat exchanger, diesel generator coolers, HVAC chiller, and to the suction of the service water booster pumps when these components are placed in-service. Prior to this time, the flow is rapidly approaching required flow and sufficient pressure is developed as valves finish their stroke. Each of the above-listed components will be starting to perform their accident mitigation function, either directly or indirectly depending upon the use of the component, and will be operational within the service water response time of 71.5/81.5 seconds[1/]. Only the service water booster pumps have a direct impact on the accident analysis via the RBCUs' heat removal capability as discussed below.

[1/] Total time is 1.5 second instrument response after setpoint is reached, plus 10 seconds diesel generator start, plus 10 seconds to reach service water pump start and begin 3116-SW opening via Engineered Safety Features Loading Sequencer, plus 60 seconds stroke time for 3116-SW. During this total time, the service water pumps start and the service water pump discharge valve begins to open at 11.5 seconds and the pump discharge valve is fully open at 71.5 seconds without a diesel generator start required and 21.5 seconds and 81.5 seconds including a diesel generator start.

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

The RBCU response time includes: 1) the start of the RBCU fan and the service water booster pumps and, 2) all the service water valves which must be driven to the fully opened or fully closed position. This condition of the valves allows the flow to become fully established through the RBCU. Prior to this time, the flow is rapidly approaching required flow as the valves finish their stroke. Although the RBCU would be removing heat throughout the Engineered Safety Features response time, the accident analysis does not assume heat removal capability from 0 to 71.5 seconds [2/] because the industrial cooling water system is not completely isolated until 71.5 seconds. A linear ramp increase from 95% full heat removal capability to 100% full heat removal capability is assumed by the accident analysis to start at 71.5 seconds and end at 86.5 seconds[3/]. Full heat removal capability is assumed at 86.5 seconds based on the position of the valve 3107-SW.

2/ Total time is 1.5 second instrument response after setpoint Is reached, plus 10 second diesel start plus 60 seconds* for valves to isolate industrial cooling water system.

3/ Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel generator start plus 75 seconds to stroke valves 3107A, B-SW.

* During this time period, the Engineered Safety Features Loading Sequencer starts the RBCU fans at 25 seconds and service water booster pumps at 30 seconds after the valves begin to stroke.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 146 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 April 6, 2000, the South Carolina Electric & Gas Company (the licensee), acting for itself and as agent for South Carolina Public Service Authority, submitted a request to amend the Technical Specifications (TS) for the Virgil C. Summer Nuclear Station. The purpose of the proposed changes is to eliminate response time testing (RTT) of the Engineered Safety Feature Actuation System (ESFAS) and the Reactor Trip System (RTS) either by measuring or by verifying the response time for specific components in the systems. The proposed changes will affect the following TS:

a. Definitions 1.12, Engineered Safety Feature Response Time, and 1.26, Reactor Trip System Response Time

b. Surveillance Requirements 4.3.1.2 and 4.3.2.2

c. Bases Sections B 3/4.3.1 and B 3/4.3.2

The current definitions for Engineered Safety Feature Response Time and Reactor Trip System Response Time imply that the response time must be measured; the proposed changes will permit verification of the response times. The proposed changes to Surveillance Requirements 4.3.1.2 and 4.3.2.2 will replace the words "demonstrated" and "tested" by the words "verified" and "verification," respectively. The proposed changes to the Bases Sections B 3/4.3.1 and B 3/4.3.2 will reflect the above-mentioned changes.

The proposed TS changes basically involve two separate requests. The first change is to eliminate periodic sensor RTT in accordance with Westinghouse Proprietary Class 2C Topical Report WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," pre-approved August 1995, and approved January 1996. The second change is to eliminate protection channel RTT in accordance with WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," approved October 6, 1998.

2.0 EVALUATION

In approving WCAP-13632, Revision 2, the NRC staff concluded that any sensor failure that can significantly degrade senor response time can be detected during the performance of other surveillance tests, principally calibration. In approving WCAP-13632, Revision 2, the NRC staff also stipulated four conditions that a licensee must meet when submitting plant-specific amendment requests. The licensee's responses to these conditions are as follows:

1. NRC's Condition: Perform a hydraulic RTT prior to installation of a new transmitter/switch or following refurbishment of the transmitter/switch (e.g., sensor cell or variable damping components) to determine an initial sensor-specific response time value.

   Licensee's Response: Consistent with the proposed TS changes (including the associated Bases for 4.3.1.2 and 4.3.2.2) and EPRI Report NP-7243, Revision 1, the applicable plant procedures will include revisions which stipulate that pressure sensor response times must be verified by performance of an appropriate response time test prior to placing a sensor into operational service and re-verified following maintenance that may adversely affect sensor response time.

   NRC Staff's Review: The staff finds the licensee's response acceptable.

2. NRC's Condition: For transmitters and switches that use capillary tubes, perform an RTT after initial installation and after any maintenance or modification activity that could damage the capillary tubes.

   Licensee's Response: Plant procedure revisions (and/or other administrative controls) will stipulate that pressure sensors (transmitters and switches) utilizing capillary tubes, e.g., containment pressure, must be subject to RTT after installation and following any maintenance or modification which could damage the transmitter capillary tubes.

   NRC Staff's Review: The licensee's action plan is acceptable to the staff.

3. NRC's Condition: If variable damping is used, implement a method to assure that the potentiometer is at the required setting and cannot be inadvertently changed or perform hydraulic RTT of the sensor following each calibration.

   Licensee's Response: V. C. Summer Nuclear Station [VCS] does not have any pressure transmitters with installed variable damping for any RTS or ESFAS application which requires RTT. No plant procedures or administrative controls will require revision as a result. Should VCS replace any transmitters in the future with ones having variable damping capability, at that time procedure changes and/or administrative controls will be enacted to assure the variable damping potentiometer cannot be inadvertently changed.

   NRC Staff's Review: The staff finds the licensee's response acceptable.

4. NRC's Condition: Perform periodic drift monitoring of all Model 1151, 1152, 1153, and 1154 Rosemount pressure and differential pressure transmitters, for which RTT elimination is proposed, in accordance with the guidance contained in Rosemount

Technical Bulletin No. 4 and continue to remain in full compliance with any prior commitments to Bulletin 90-01, Supplement 1, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," dated December 22, 1992. As an alternative to performing periodic drift monitoring of Rosemount transmitters, licensees may complete the following actions: (1) ensure that operators and technicians are aware of the Rosemount transmitter loss of fill-oil issue and make provisions to ensure that technicians monitor for sensor response time degradation during the performance of calibrations and functional tests of these transmitters, and (2) review and revise surveillance testing procedures, if necessary, to ensure that calibrations are being performed using equipment designed to provide a step function or fast ramp in the process variable and that calibrations and functional tests are being performed in a manner that allows simultaneous monitoring of both the input and the output response of the transmitter under test, thus allowing, with reasonable assurance, the recognition of significant response time degradation.

Licensee's Response: VCS provided responses to NRC Bulletins 90-01 and 90-01, Supplement 1, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount" by letters dated July 19, 1990, May 8, 1992, and February 23, 1993. These letters address the position and actions taken with respect to the loss of fill-oil for the Rosemount transmitters.

NRC Staff's Review: By letter dated February 13, 1995, the NRC concluded that the licensee had addressed the NRC's requirements to NRC Bulletin 90-01, Supplement 1, satisfactorily and the NRC staff finds the licensee's response acceptable.

In Attachment 2 of their submittal, the licensee listed all the functions and the sensors that are affected by the proposed TS changes. The list includes Barton 752, Barton 752 with model 351 sealed sensor, Barton 763/763A, Barton 764, and Rosemount 1154 sensors. WCAP-13632-P-A, Revision 2, specifies 0.400 seconds as response time for these sensors except for Rosemount 1154 sensors, for which the WCAP does not specify any time. The licensee reviewed the past response time of all Rosemount 1154 sensors and found 0.260 seconds as the worst-case value. The licensee conservatively selected 0.400 seconds as the response time for all these sensors including Rosemount 1154 sensors and the NRC staff finds this selection acceptable.

By letters dated October 6, 1998, and November 3, 1998, the NRC approved WCAP-14036-P-A, Revision 1, sponsored by the Westinghouse Owners Group (WOG), which is for elimination of channel response time tests. In the proposed methodology, the response times will be verified by summing allocated times for sensors, the process protection system, the nuclear instrumentation system, and the logic system. These allocated times will be added to the measured times for the actuated components and compared to the overall analysis limits. In approving WCAP-14036-P-A, Revision 1, the NRC addressed the Failure Modes and Effect Analysis (FMEA), which was contained in and supported by WCAP-14036-P-A by stipulating, "Since the performance of RTT is a TS requirement, licensees referencing WCAP-14036-P-A, Revision 1, must submit a TS amendment to eliminate that requirement for the identified equipment. In that amendment request, the licensee must verify that the FMEA performed by the WOG is applicable to the equipment actually installed in the licensee's facility, and that the analysis is valid for the versions of the boards used in the protection system."

The licensee responded, "Attachment II to this letter includes Tables, which identify the equipment in the various instrument loops for specific protective functions. Note 4 contains a statement that the FMEA provided by the WOG in WCAP-14036-P-A, Revision 1, is applicable to the installed equipment in the population for which this change request is being submitted. The analysis is valid for the versions of circuit boards identified. Additional versions of circuit boards, which are not included in the approved WOG FMEA will not be included in this population. A separate FMEA will be submitted to the NRC at the time these additional boards are requested to be included in this population."

The NRC staff finds that the response times specified in Attachment 2 of the licensee's submittal conform to WCAPs 13632 and 14036.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (65 FR 25768). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The proposed RTT changes conform to the NRC-approved WOG reports WCAP-13632-P-A, Revision 2, and WACP-14036-P-A, Revision 1. The licensee satisfactorily addressed all the NRC's requirements stipulated in the safety evaluations of these two reports. The NRC staff has approved similar changes for other nuclear plants and finds the proposed changes acceptable.

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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