Docket File

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

May 20, 1999

Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 -ISSUANCE OF AMENDMENTS - REVISED CALCULATION METHODOLOGY FOR LARGE BREAK LOSS-OF-COOLANT ACCIDENT MASS AND ENERGY RELEASE ANALYSIS (TAC NOS. MA3768 AND MA3769)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 110 to Facility Operating License No. NPF-76 and Amendment No. 97 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2 (STP). The amendments authorize changes to the updated final safety analysis report (UFSAR) in response to your application dated September 29, 1998, which identified the changes as an unreviewed safety question. The amendments are being issued pursuant to the requirements of 10 CFR 50.59(c).

The amendments allow use of a revised methodology to calculate mass and energy release following a postulated large-break loss-of-coolant accident (LBLOCA) for the replacement steam generators. You intend to replace the Westinghouse Model E original steam generators currently in Units 1 and 2 with Westinghouse Delta 94 replacement steam generators in the future and determined that changes to the existing licensing basis were necessary to properly model the improved performance of the replacement steam generators. The revised methodology will more accurately model the LBLOCA subsequent to the point at which the replacement steam generators would be fully cooled and depressurized in an accident . The revised method for calculating the mass and energy release following an LBLOCA use NRC-approved methodologies and computer programs and an endorsed NRC correlation for core decay heat generation. Because approval of the amendments is based on the revised methodology being incorporated into the UFSAR for the replacement steam generators, the incorporation is a condition of the amendments and must be completed by the date the UFSAR is updated in accordance with 10 CFR 50.71(e).

The amendments do not change the Technical Specifications for the units.

DF01

CP-1

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY

Thomas W. Alexion, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 110 to NPF-76
2. Amendment No. 97 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated September 29, 1998, 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, as amended, 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 license 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.

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*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, by Amendment No. 110 , the Facility Operating License No. NPF-76 is amended to authorize revision of the Updated Final Safety Analysis Report (UFSAR) to incorporate the revised methodology to calculate mass and energy release following a postulated large-break loss-of-coolant accident for replacement steam generators as set forth in Attachment 4 of the application for amendment by STP Nuclear Operating Company dated September 29, 1998, and evaluated in the staff's safety evaluation enclosed with this amendment. STP Nuclear Operating Company shall incorporate the revisions into the next UFSAR update in accordance with the schedule in 10 CFR 50.71(e).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommisisoning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: May 20, 1999

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated September 29, 1998, 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, as amended, 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 license 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.

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*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, by Amendment No. 97 , the Facility Operating License No. NPF-80 is amended to authorize revision of the Updated Final Safety Analysis Report (UFSAR) to incorporate the revised methodology to calculate mass and energy release following a postulated large-break loss-of-coolant accident for replacement steam generators as set forth in Attachment 4 of the application for amendment by STP Nuclear Operating Company dated September 29, 1998, and evaluated in the staff's safety evaluation enclosed with this amendment. STP Nuclear Operating Company shall incorporate the revisions into the next UFSAR update in accordance with the schedule in 10 CFR 50.71(e).

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: May 20, 1999

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 110 AND 97 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By application dated September 29, 1998, STP Nuclear Operating Company (the licensee) requested changes to the licenses for South Texas Project, Units 1 and 2 (STP) to authorize changes to the updated final safety analysis report (UFSAR). The proposed changes would allow use of a revised methodology to calculate mass and energy release following a postulated large-break loss-of-coolant accident (LBLOCA) for the Westinghouse Delta 94 replacement steam generators (RSGs). The STP has pressurized-water reactors (PWR)..

The licensee intends to remove the current Westinghouse Model E original steam generators (OSGs) from Units 1 and 2, and to install the RSGs in the future. The licensee has determined that changes to the existing licensing basis are necessary to more accurately model the improved performance of the RSGs during the LBLOCA subsequent to the point at which the steam generators are fully cooled and depressurized for the RSGs. The revised method for calculating the mass and energy release following an LBLOCA will use NRC-approved methodologies and computer programs and will apply an endorsed NRC correlation for decay heat generation.

The licensee has determined that this revision to a methodology for steam generators described in the UFSAR constitutes an unreviewed safety question (USQ) that requires prior NRC review and approval. The revised methodology was submitted in the letter of September 29, 1998.

2.0 DISCUSSION

Performance improvements incorporated within the RSG design require evaluations or analyses of design-basis accidents that depend on the following considerations:

- Reactor coolant system (RCS) volume, and flow resistance
- Steam generator (SG) volume, metal mass, heat transfer surface area, and heat transfer coefficient

The RSGs will increase the RCS primary system liquid volume by approximately 9.5 percent (based on nominal values) and thereby increase the RCS mass. A reduction in the reactor vessel upper head temperature to that of the cold leg inlet temperature value also increases the RCS mass. These effects combine to increase the RCS liquid mass by approximately 7.7 percent. The secondary side initial water mass increases by approximately 11 percent for each RSG. These factors combine to create an adverse effect on the containment pressure and temperature response to a LOCA. To address these issues, the licensee proposes to use an improved methodology for the determination of the LOCA mass and energy releases and the consequent containment pressure and temperature response. The LBLOCA is a subset of LOCAs and would provide the most severe consequences for the containment.

The analysis of the containment pressure and temperature response resulting from an LBLOCA is typically performed in two parts. The first part determines the mass and energy release from the RCS. The second part uses this mass and energy release to determine the resultant pressure and temperature (P/T) response in containment. P/T results from this analysis are then used to determine:

- If systems, structures, and components important to safety are bounded by equipment qualification limits,
- The parameters for Integrated Leak Rate Testing, such as peak containment pressure (P_a), and
- If hydrogen generation rate caused by elevated containment temperatures is less than hydrogen recombiner capacity.

2.1 <u>LOCA Mass and Energy Releases</u>

The LOCA mass and energy release analysis calculation model is typically divided into four phases: (1) blowdown, (2) refill, (3) reflood, and (4) "froth boiling" or post reflood. "Froth boiling" is a consideration only for the case of a double-ended pump suction guillotine (DEPSG) break, otherwise this last phase is referred to as "post reflood." These phases represent periods within the accident.

(1) Blowdown Phase: The period of time commencing with accident initiation with the reactor at full power, steady-state operation, and ending when the RCS and containment reach pressure equilibrium. Typically, the reactor is drained of coolant during the blowdown phase, which is usually ≤30 seconds.

(2) Refill Phase: The period of time commencing with the end of the blowdown phase, and ending when the emergency core cooling system (ECCS) injection flow has filled the lower reactor vessel plenum to the bottom of the reactor core.

(3) Reflood Phase: The period of time commencing with the end of the refill phase and ending when the reactor core is covered by the ECCS effluent, quenching the core.

(4) Post-Reflood Phase: The period of time commencing with the end of the reflood phase and continuing as long as significant energy is being released from the SGs into containment. In the case of a pump suction break, the initial portion of this phase commencing with the

end of the reflood phase and ending when all SGs are depressurized to the containment design pressure is referred to as the "froth boiling phase." Subsequent to the "froth boiling phase," the remainder of the time during which heat continues to be released from the SGs is referred to as "post-reflood phase."

The current licensing basis for STP uses the approved methodology described in WCAP-8264-P-A, "Westinghouse Mass and Energy Release Data for Containment Design," Revision 1, dated 1974 (Proprietary information. Not publicly available.), to determine the mass and energy releases up to the point of cold leg switchover. After cold leg switchover, the mass and energy releases are determined using the reactor vessel model in the COPATTA computer code.

The licensee proposes to use the methodology described in WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model For Containment Design March 1979 Version," dated May 1983 (Proprietary information. Not publicly available.), for the LOCA mass and energy release calculations associated with the RSGs for the time period prior to the point at which the SGs are cooled and depressurized. In this model, the SGs are cooled and depressurized to atmospheric pressure approximately 1 hour after initiation of the LOCA event. The NRC has approved the methodology described in WCAP-10325-P-A for such applications in the letter from C.E. Rossi (NRC) to W.J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10325," dated February 17, 1987.

2.2 Containment Response

The current licensing basis approved methodology for the STP containment pressure and temperature response analysis to a LOCA is for the use of the COPATTA computer code during the entire transient.

For the analysis of the containment pressure and temperature response for the RSGs, the licensee has proposed to use the CONTEMPT4/MOD5 computer code (Lin, C.C., Economos, C., Lehner, J.R., and Ng, K.K., "CONTEMPT4/MOD4—A Multicompartment Containment System Analysis Program," Brookhaven National Laboratory, March 1984, NUREG/CR-3716, BNL-NUREG-51754, and Lin, C.C., "CONTEMPT4/MOD5—An Improvement to CONTEMPT4/MOD4 Multicompartment Containment System Analysis Program," Brookhaven National Laboratory, September 1984, NUREG/CR-4001, BNL-NUREG-51824). The CONTEMPT4/MOD5 computer code was developed for the NRC to be used for the analysis of containment pressure and temperature response during postulated design-basis accidents, such as an LBLOCA. This computer code, and its associated methodology, have been reviewed and accepted by the NRC for use by the licensee for the analysis of the containment pressure and temperature response to steam line break events, as discussed in Section 2.6.1 of the safety evaluation for the V5H fuel upgrade in Amendments 61 and 50 to STP in the staff's letter dated May 27, 1994.

CONTEMPT4/MOD5 has also been used by other licensees to perform containment pressure and temperature response analyses to LOCAs. It has also been verified under the licensee's quality assurance program for the analysis of LBLOCAs. This verification included benchmarks against the COPATTA computer code. Results from the licensee's benchmarking of the revised mass and energy release methodology proposed in this submittal exhibited good agreement between the CONTEMPT4/MOD5 and COPATTA codes.

2.3 Description of the LOCA Mass and Energy Calculation for RSGs

The WCAP-10325-P-A methodology uses the SATAN-VI computer code (instead of the SATAN-V computer code used in the current WCAP-8624-P-A methodology) to calculate the LOCA mass and energy release during the blowdown phase. The refill phase is assumed to take zero time, consistent with the guidance provided in Standard Review Plan (SRP) 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," II.3.c, "PWR Core Reflood Phase (Cold Leg Break Only)," which results in a conservative mass and energy release calculation. The reflood phase uses an improved version of the WREFLOOD computer code to determine mass and energy release, as described in WCAP-10325-P-A. The FROTH computer code is used to model the post-reflood phase of the transient for both the current and the proposed mass and energy release methodology. The FROTH code calculates the heat release rates from any two-phase mixture present in the SG tubes.

The secondary-side energy is removed in stages from the SGs. The FROTH code is used to calculate the heat removal from the secondary mass until the secondary temperature reaches the saturation temperature (T_{sat}) for containment design pressure. After this point, for both the current and the proposed mass and energy release methodology, the EPITOME code is used to continue the post-reflood phase calculations for the SG cooldown instead of the FROTH code. The EPITOME code is a Westinghouse-developed computer code version of hand calculations previously used by Westinghouse for design analysis in this region. The EPITOME code has been used in similar submittals approved by the NRC. The SG secondary energy is removed in two stages. The first stage uses an energy removal rate until the SG reaches T_{sat} at the user-specified intermediate equilibration pressure, at which point the secondary pressure is assumed to have reached actual containment pressure. The second stage uses an energy removal rate until final depressurization, when the secondary temperature reaches T_{sat} of 212 °F at 14.7 psia. The broken loop heat removal and the intact loop SG heat removal are calculated separately.

To account for the containment sump enthalpy, the FROTH and EPITOME analyses are provided with an estimate of the containment sump temperature as input to be used in the calculation of the mass and energy release from the reactor containment system.

The containment pressure and temperature analysis is then performed using the mass and energy releases as input to the CONTEMPT4/MOD5 code. The resultant CONTEMPT4/MOD5 sump temperature is then compared to the estimate of the containment sump temperature provided to the FROTH and EPITOME analyses. If the calculated sump temperature is lower than the temperature estimated in the first step, then the analysis has produced conservative results. Repeating the analysis with the newly calculated containment sump temperature as input to the first step can be used to refine the outcome.

In the current methodology (WCAP-8264-P-A) for the OSGs and in the proposed methodology (WCAP-10325-P-A) for the RSGs, it is possible to use the EPITOME analysis to the end of the transient. However, to ensure a conservative calculation during the period prior to the SGs being cooled and depressurized to T_{sat} at 14.7 psia, the current methodology assumes a containment sump temperature higher than that actually calculated by the CONTEMPT4/MOD5 containment computer code. This introduces an artificial energy source into containment during the remaining time.

The proposed core decay heat model discontinues the use of the EPITOME code after the SGs are cooled and depressurized to T_{sat} at 14.7 psia, to eliminate this artificial energy source, and starts to add core decay heat to containment. The core decay heat value used in the proposed methodology is calculated using the Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling," decay heat correlation as described in SRP Section 9.2.5, "Ultimate Heat Sink." The RCS effluent superheated by decay heat is transformed to steam, using a pressure-flash model, and its mass and energy is added directly to the containment atmosphere without mixing with ECCS injection water. The use of the ASB 9-2 decay heat model and the treatment of the steam are consistent with the guidance provided in SRP 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," II.3.e, "PWR Decay Heat Phase." The mass and energy of saturated and subcooled liquid is added directly to containment sump. The long-term containment pressure and temperature performance using this method has been shown by the licensee to be consistent with COPATTA code results.

2.4 UFSAR Update

The licensee stated that the STP UFSAR is to be revised to incorporate a new section 6.2A.1, "Containment Functional Design." This section addresses the reactor containment analyses for the RSGs. Section 6.2.1 of the UFSAR describes the reactor containment analyses for the OSGs. Analyses or references that apply to either SG are included in Section 6.2 only.

The sources of energy are described in the UFSAR in Section 6.2A.1.3.2, "Energy Sources," and Section 6.2A.1.3.7, "Metal/Water Reaction," consistent with the guidance of SRP 6.2.1.3, II.B.1. Section 6.2A.1.1.1, "Design Bases," describes the break size and location, as well as the effects of the engineered safety feature systems on energy removal, consistent with SRP 6.2.1.3, II.B.2. Section 6.2A.1.3, "Mass and Energy Release Analyses For Postulated Loss-of-Coolant Accidents," describes the calculations for the mass and energy releases from a LOCA consistent with SRP 6.2.1.3, II.B.3. The initial blowdown phase, PWR core reflood phase and PWR post-reflood phase are evaluated with the approved methodologies described in WCAP-10325-P-A. Section 6.2A.1.3.8, "Energy Inventories," provides the results of the energy balance for the primary and secondary systems for the time period up to the time the SGs are cooled and depressurized to a saturation temperature of 212 °F at 14.7 psia, consistent with SRP 6.2.1.3, Section III, "Review Procedures."

The PWR decay heat phase is evaluated with the STP decay heat model. The core decay heat value used in the proposed methodology is calculated using the Branch Technical Position ASB 9-2 decay heat correlation as described in SRP Section 9.2.5. The RCS effluent superheated by decay heat is transformed to steam, using a pressure-flash model, and its mass and energy is added directly to the containment atmosphere without mixing with ECCS injection water. The use of the ASB 9-2 decay heat model and the treatment of the steam are consistent with the guidance provided in SRP 6.2.1.3, II.3.e. Mass and energy of saturated and subcooled liquid is added directly to containment sump.

UFSAR Section 6.2A.1.1.3, "Design Evaluation," describes the containment peak pressure and temperature analysis including a description of the CONTEMPT4/MOD5 model used for the evaluation of the containment response resulting from LOCAs with installed RSGs.

3.0 EVALUATION

The staff has reviewed the licensee's submittal to allow the use of a revised methodology to calculate mass and energy release following a postulated LBLOCA. The licensee intends to replace the OSGs in Units 1 and 2 with the RSGs. The revised method for calculating the mass and energy release following an LBLOCA will be used to support the installation of the RSGs. The licensee has determined that this revised method of calculation that was previously described in the UFSAR constitutes a USQ that requires NRC review and approval. The staff agrees with the licensee's determination based on 10 CFR 50.59(c).

The revised method uses NRC-approved methodologies and computer programs and an endorsed NRC correlation for decay heat generation.

3.1 LOCA Mass and Energy Releases

The LOCA mass and energy releases for the initial blowdown phase, PWR core reflood phase, and PWR post-reflood phase are evaluated with the methodologies described in WCAP-10325-P-A. WCAP-10325-P-A has been previously reviewed and accepted by the staff for this purpose. The staff finds this methodology to be acceptable for use by the licensee for the calculation of the LOCA mass and energy releases with the RSGs.

The PWR decay heat phase is evaluated with the decay heat model. The core decay heat value used in the revised methodology is calculated using the Branch Technical Position ASB 9-2. The RCS effluent superheated by decay heat is transformed to steam, using a pressure-flash model, and its mass and energy is added directly to the containment atmosphere without mixing with ECCS injection water. The use of the ASB 9-2 decay heat model and the treatment of the steam are consistent with the guidance provided in SRP 6.2.1.3. Mass and energy of saturated and subcooled liquid is added directly to containment sump. The staff has reviewed the decay heat model for evaluating the PWR decay heat phase of the LOCA for use with the RSGs and concludes that it is acceptable.

3.2 Containment Response

The analysis of the containment pressure and temperature response for installed RSGs to LOCAs uses the CONTEMPT4/MOD5 computer code. This computer code, and its associated methodology, has been reviewed and accepted by the NRC for use by the licensee for the analysis of the containment pressure and temperature response to steam line break events. The CONTEMPT4/MOD5 code has also been used by other licensees to perform containment pressure and temperature response analyses to LOCAs. It has also been verified under the STP quality assurance program for the analysis of LBLOCAs. This verification included benchmarks against the COPATTA code. The CONTEMPT4/MOD5 code has been previously reviewed and accepted by the staff for this purpose (see Section 2.2 above). The staff finds this methodology to be acceptable for use by the licensee for the calculation of the containment pressure and temperature response to LOCA mass and energy releases with the RSGs.

3.3 UFSAR Update

In addition to reviewing the revised LBLOCA methodology for the RSGs, it is necessary to have the revised methodology incorporated into a licensee-controlled document where future changes to the methodology are controlled by the regulations or the administrative section of the Technical Specifications. The licensee has proposed to incorporate the revised methodology into the UFSAR where changes to the methodology will be controlled by the regulations in 10 CFR 50.59. The staff concludes that the change control criteria in 10 CFR 50.59 are acceptable to control any future change to the revised LBLOCA methodology for the RSGs.

The staff has reviewed the UFSAR Section 6.2A.1 updates which describe the reactor containment analysis for the RSGs. The updates describe the methodology used for the mass and energy release calculations based on the WCAP-10325-P-A and decay heat model. The updates describe the methodology used for the containment peak pressure and temperature calculations based on the use of the CONTEMPT4/MOD5 code. The LOCA break spectrum, the effects of the engineered safety feature systems on the mass and energy releases, and the energy sources and initial conditions used to maximize the calculated peak containment pressure and temperature are also described. The staff concludes that the Section 6.2A.1 UFSAR updates are acceptable for the RSGs.

3.4 Conclusion

Based on the above, the staff concludes that the proposed amendments are acceptable. Because approval of the amendments is based on the revised methodology for the RSGs being incorporated into the UFSAR, the incorporation is a condition of the amendments and must be completed by the next UFSAR update in accordance with the schedule in 10 CFR 50.71(e).

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 64123, November 18, 1998). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: E. Throm

Date: May 20, 1999