March 2, 2000

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 ON RADIOLOGICAL ASPECTS OF STEAM GENERATOR OPERATION AT REDUCED FEEDWATER TEMPERATURE AND OPERATION WITH REPLACEMENT STEAM GENERATORS (TAC NOS. MA3820 AND MA3821)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. NPF-76 and Amendment No. 112 to Facility Operating License No. NPF-80 for the South Texas Project (STP), Units 1 and 2, respectively, in response to your application dated September 30, 1998, as supplemented by letters dated May 14 and October 21, 1999. The amendments revise the STP, Units 1 and 2, offsite dose licensing bases to account for (1) operation of the existing steam generators at reduced feedwater inlet temperatures and (2) operation with the new replacement steam generators, also at a reduced feedwater temperature. The changes revised calculated offsite doses for four existing Updated Final Safety Analysis Report Chapter 15 accidents and added a discussion in Chapter 15 of the radiological analysis for the voltage-based criteria for steam generator tubes.

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,

/RA/

John A. Nakoski, Senior 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. 124 to NPF-76
2. Amendment No. 112 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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cc:

Mr. Cornelius F. O'Keefe
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 910
Bay City, TX 77414

A. Ramirez/C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

Mr. M. T. Hardt
Mr. W. C. Gunst
City Public Service Board
P. O. Box 1771
San Antonio, TX 78296

Mr. G. E. Vaughn/C. A. Johnson
Central Power and Light Company
P. O. Box 289
Mail Code: N5012
Wadsworth, TX 74483

INPO
Records Center
700 Galleria Parkway
Atlanta, GA 30339-3064

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

D. G. Tees/R. L. Balcom
Houston Lighting & Power Co.
P. O. Box 1700
Houston, TX 77251

Judge, Matagorda County
Matagorda County Courthouse
1700 Seventh Street
Bay City, TX 77414

A. H. Gutterman, Esq.
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, DC 20036-5869

Mr. J. J. Sheppard, Vice President
Engineering & Technical Services
STP Nuclear Operating Company
P. O. Box 289
Wadsworth, TX 77483

S. M. Head, Supervisor, Licensing
Quality & Licensing Department
STP Nuclear Operating Company
P. O. Box 289
Wadsworth, TX 77483

Office of the Governor
ATTN: John Howard, Director
    Environmental and Natural
    Resources Policy
P. O. Box 12428
Austin, TX 78711

Jon C. Wood
Matthews & Branscomb
One Alamo Center
106 S. St. Mary's Street, Suite 700
San Antonio, TX 78205-3692

Arthur C. Tate, Director
Division of Compliance & Inspection
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756

Jim Calloway
Public Utility Commission of Texas
Electric Industry Analysis
P. O. Box 13326
Austin, TX 78711-3326

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
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 the City of Austin, Texas (COA) (the licensees), dated September 30, 1998, as supplemented by letters dated May 14 and October 21, 1999, 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 the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, Facility Operating License No. NPF-76 is hereby amended to authorize changes to the Updated Final Safety Analysis Report (UFSAR) to reflect the description of the revised calculated offsite doses for four existing UFSAR Chapter 15 accidents as set forth in the application for amendment by the licensee dated September 30, 1998, as supplemented by letters dated May 14 and October 21, 1999.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Date of Issuance: March 2, 2000

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
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 the City of Austin, Texas (COA) (the licensees), dated September 30, 1998, as supplemented by letters dated May 14 and October 21, 1999, 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.

---

*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 the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, Facility Operating License No. NPF-80 is hereby amended to authorize changes to the Updated Final Safety Analysis Report (UFSAR) to reflect the description of the revised calculated offsite doses for four existing UFSAR Chapter 15 accidents as set forth in the application for amendment by the licensee dated September 30, 1998, as supplemented by letters dated May 14 and October 21, 1999.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Date of Issuance: March 2, 2000

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 124 AND 112 TO

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

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated September 30, 1998, as supplemented by letters dated May 14 and October 21, 1999, the STP Nuclear Operating Company, et al. (the licensee), submitted a request for changes to the South Texas Project (STP), Units 1 and 2, licensing basis. The purpose of the license change was to revise the offsite dose licensing bases to account for (1) operation of the existing steam generators at reduced feedwater inlet temperatures and (2) operation with the new replacement steam generators, also at a reduced feedwater temperature. The changes proposed by the licensee involve revised calculated offsite doses for four existing Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents. Also, the licensee provided for the staff's information a discussion that is to be included in Chapter 15 of the radiological analysis for the voltage-based criteria for steam generator tubes. The May 14 and October 21, 1999, supplements provided additional clarifying information that was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination.

In addition to the changes precipitated by the reduced feedwater temperature, the licensee also chose to update the methodologies associated with its accident analyses and to improve the analytical assumptions. As a result of these changes, the offsite dose consequences of three of the accidents increased and required review as unreviewed safety questions (USQs). The offsite consequences of the main steamline break (MSLB), the rod cluster control assembly ejection accident, and the locked rotor increased. The decrease in feedwater temperature would result in a reduction in the dose consequences of a steam generator tube rupture (SGTR) event. Consequently, the three accidents, which resulted in an increase in dose consequences, required NRC review as USQs while the accident that resulted in a decrease in consequences required staff review due to a change in critical analysis assumptions.

2.0 EVALUATION

The current STP licensing basis calculations for offsite dose consequences from UFSAR Chapter 15 analyzed accidents are based upon the assumption of operation with existing

Westinghouse Model E steam generators at a normal feedwater temperature of 440 ^{o}F. However, due to continued degradation of the steam generator tubes, the licensee proposed to operate with the existing Model E steam generator feedwater inlet temperatures as low as 420 ^{o}F in order to achieve 100 percent reactor power during degraded steam generator conditions. Eventually, the existing Model E steam generators will be replaced with the Westinghouse Model Δ94 steam generators. The replacement steam generators will operate with feedwater inlet temperature as low as 390 ^{o}F.

The licensee calculated the doses to individuals located offsite at the exclusion area boundary (EAB) and the low-population zone (LPZ) to demonstrate that the offsite doses would meet the limitations of 10 CFR Part 100 or some fraction thereof, depending upon the accident. The licensee did not perform an assessment of the control room operator doses to determine whether the consequences of the above accidents would be within the guidelines of General Design Criterion (GDC) 19 of Appendix A of 10 CFR Part 50 because it had concluded that such an analysis was not required since the loss-of-coolant accident (LOCA) is the limiting accident for the control room and technical support center doses.

The accidents for which the licensee performed Chapter 15 reanalyses and the NRC guidelines for confirming 10 CFR Part 100 limits for offsite doses are not exceeded as specified in the Standard Review Plan (SRP) for these accidents are as follows:

1. Main Steam Line Break (guidelines apply to events both with and without alternate repair criteria conditions)

    - pre-existing spike case - 25 rem whole body, 300 rem thyroid
    - accident-initiated spike case - 2.5 rem whole body, 30 rem thyroid
    - with fuel damage - 25 rem whole body, 300 rem thyroid

2. Steam Generator Tube Rupture

    - pre-existing spike case - 25 rem whole body, 300 rem thyroid
    - accident-initiated spike case - 2.5 rem whole body, 30 rem thyroid

3. Locked Rotor

    - 2.5 rem whole body, 30 rem thyroid
    - with fuel damage - 25 rem whole body, 300 rem thyroid

4. Rod Ejection

    - 6 rem whole body, 75 rem thyroid

NRC guidelines for confirming GDC 19 criteria for control room doses are not exceeded as specified in Section 6.4 of the SRP are 5 rem whole body and 30 rem thyroid.

For the locked rotor, rod ejection, MSLB, and SGTR, the staff reviewed the licensee's assessments and performed confirmatory calculations. The staff calculated doses for individuals located offsite at the EAB and at the LPZ. The staff's calculations included doses for the control room operators to confirm regulatory limits would not be exceeded. The staff's calculated control room doses were well below the GDC 19 guidelines for control room personnel and the staff concluded it was reasonable for the licensee to conclude that LOCA is the limiting accident for doses to control room personnel for conditions where voltage-based repair criteria for steam generator tubes are not applied. For those conditions that are analyzed with the voltage-based repair criteria for steam generator tubes applied, the MSLB is limiting. Section 2.1 provides the staff's assessment of the potential consequences on postulated accidents of the decrease in feedwater temperature.

The licensee also assessed the consequences of the feedwater temperature reduction on other accidents such as the LOCA, waste gas decay tank release, the feedwater system pipe break, reactor coolant pump shaft break, small line break outside containment, and liquid radwaste tank processing system failure and on primary side and secondary side isotopic quantities. Based on its review of the analysis performed by the licensee, the staff concluded that the effect of reduced feedwater temperatures on these other accidents was negligible.

2.1 Accidents Analyzed

The licensee's assessment of the consequences of a postulated accident affected by a reduction in feedwater temperature involved two cases. One case assessed the impact of the reduced feedwater temperature (420 ^{o}F) for the existing Model E steam generators. The second case involved the impact of the replacement steam generators (the Model Δ94), which incorporate a feedwater temperature of 390 ^{o}F. The licensee indicated that, for completeness, it had included the previously submitted and NRC accepted revised MSLB accident analysis that incorporated the STP's utilization of the alternate repair criteria (ARC) for the Model E steam generators. Since the MSLB with ARC was accepted by the NRC (see the May 22, 1996, safety evaluation for Amendment 83 to STP Unit 1's Technical Specifications and the September 24, 1998, safety evaluation for Amendment 83 to STP Unit 2's Technical Specifications), it was not necessary for the staff to review the radiological aspects of the MSLB with ARC here.

2.1.1 Main Steam Line Break Without Voltage-Based Criteria

The licensee assessed the effects of the consequences of operation with the Model Δ94 steam generators and operation with the Model E steam generators with a lower feedwater temperature. The licensee indicated that the difference in model parameters between the current Model E steam generators with a feedwater temperature of 440 ^{o}F and the Model E steam generators at 420 ^{o}F and the Model Δ94 steam generators at a feedwater temperature of 390 ^{o}F is slight with the impact upon doses small. The licensee indicated that in the event of an MSLB, the fuel does not experience departure from nucleate boiling (DNB); therefore, fuel clad damage was not considered in the analysis of dose consequences. This was a change from the assumptions in Section 15.1.5.3 of the existing UFSAR analysis that assumes 5-percent fuel failure. However, the licensee had previously determined, and the staff had accepted (see the May 22, 1996, safety evaluation supporting Amendment 83 to STP, Unit 1's Technical Specifications), that no fuel damage would occur in the event of an MSLB accident. This was incorporated into UFSAR Section 15.1.5.1. Therefore, the staff finds this change in assumptions acceptable for the radiological aspects of the MSLB accident analysis.

The licensee indicated that the small change in dose consequences for an MSLB results more from the change in accident source term release modeling methodology rather than due to the decrease in the feedwater temperature. One change in source term release methodology involved the assumption for releases from the steam generator with the steamline break (faulted steam generator). The existing UFSAR analyses assumes primary to secondary leakage in the faulted steam generator is diluted with the water volume of the faulted steam generator (using the nominal at power water volume of the Model E steam generators). However, in the licensee's revised analyses, the licensee assumed that primary to secondary leakage in the faulted steam generator instantly flashes to steam and is immediately released to the environment undiluted by any water volume. This results in a larger release to the environment.

Another change in the licensee's modeling of the MSLB accident was associated with the release of radioactivity from the primary side to the secondary side. The existing ARC voltage-based repair criteria analysis modeled the release of isotopes from the primary side by integrating the release from 0 to 8 hours and instantaneously releasing this total amount of activity at time zero. The licensee's revised analysis used the release rate of radioactivity from the core to the primary side and assumed instantaneous mixing in the reactor coolant. The release of radioactivity to the environment was then based upon the primary to secondary leak rate.

Another major change in the licensee's revised analysis involved the use of the International Commission on Radiological Protection (ICRP) 30 dose conversion factors. Previous analyses had utilized ICRP 2 or Regulatory Guide 1.109 ("Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I") dose conversion factors. In addition, the revised analyses also incorporated a diffuse source atmospheric dispersion factor (χ/Q) for releases from the power-operated relief valves (PORVs) due to the proximity of the PORV, which is located in the isolation valve cubicle, to the control room intake. Such a dispersion factor had been utilized in previous ARC analyses. The licensee stated that the use of the diffuse source χ/Q value was appropriate.

The licensee's analysis also incorporated a release source which is unique to the STP plant. This source was leakage that could occur from the seat drains above the main steam isolation valves (MSIVs). The MSIV seat drain line valves are utilized for the removal of accumulated condensate thereby protecting the turbine from water-induced damage and to prevent water hammer in the steam lines during startup operations. During normal operation, manual valves isolate the MSIV seat drain lines. Due to the use of restricting orifices, flow from the lines is limited and no operator action is required to close the MSIV seat drain line isolation valves. However, these valves result in an additional steam release pathway that the licensee considered in its radiological analysis. This leakage is assumed to occur for 36 hours for the intact steam generators. No leakage is assumed to come from this source for the faulted steam generator because of the initial assumption that all primary to secondary leakage to the faulted steam generator is released immediately to the environment.

For the MSLB, primary coolant is assumed to be released to secondary coolant of both the intact and faulted steam generators at the technical specification primary to secondary leak rate limit. As noted above, the licensee assumes that all of the primary to secondary leakage to the faulted steam generator is released immediately to the environment with a partition factor of 1. In addition, the licensee assumes that primary to secondary leakage continues for a period of 36 hours for the MSLB analysis without ARC and for a period of 8 hours for the MSLB with the ARC analysis.

For both the existing and revised analyses, the licensee assumes that the initial secondary side activity level of dose equivalent ^{131}I is at the technical specification limit of 0.1 µCi/g. The proposed and the existing analyses both assume an iodine partition factor of 1 for iodine release from the faulted steam generator.

The staff's evaluation of the MSLB accident addressed only the consequences associated with the Model Δ94 steam generators and, as noted above, did not address the consequences associated with the ARC amendments. The assessment of the Model Δ94 steam generators involved two different analyses. One analysis assumed the accident occurred following an iodine spike, referred to as the pre-existing spike case. The second analysis assumed that the MSLB resulted in the initiation of an iodine spike, referred to as the accident-initiated spike case.

The staff assumed for both analyses a primary to secondary leak rate of 1 gpm. For the pre-existing spike case, it was assumed that primary coolant was at the maximum allowable technical specification value of dose equivalent ^{131}I at 80 percent or greater of thermal rated power. That value is 60 µCi/gm for STP. For the accident-initiated or concurrent iodine spike analysis, primary coolant activity was assumed to be at the existing 100-hour technical specification limit of dose equivalent ^{131}I for STP, 1 µCi/gm. For the accident-initiated spike case, concurrent with the MSLB, an iodine spike was assumed to occur which releases iodine from the fuel gap to the reactor coolant at a rate that is 500 times the iodine release rate to maintain primary coolant at the 100-hour technical specification limit of dose equivalent ^{131}I. For both analyses the secondary system activity was assumed to be at the technical specification limit of 0.1 µCi/gm dose equivalent ^{131}I.

The staff performed confirmatory calculations of the radiological consequences of an MSLB accident. Table 2.1.1-1 presents the assumptions utilized by the staff in its assessment. The potential consequences of an MSLB accident are presented in Tables 5-1 through 5-3. The staff's calculations show that the thyroid doses for the EAB and LPZ are within the guidelines in the SRP (see page 2) for this accident, and the doses remain below the 10 CFR Part 100 dose criteria for offsite consequences. Also, the staff's calculated doses to control room personnel were below the GDC 19 criteria for control room doses. Therefore, the staff concluded that the radiological consequences associated with the MSLB were acceptable.

2.1.2 Steam Generator Tube Rupture

The major impact of the decrease in feedwater temperature on the consequences of an SGTR accident is an increase in the initial mass in the steam generators which increases the final water volume in the ruptured steam generator. This increase in the initial quantity of steam generator mass lowers the offsite consequences since it ultimately leads to a slightly reduced steaming rate from steam generators. The licensee indicated that the steam releases for the reduced feedwater temperature cases for the Model E steam generators are bounded by the existing UFSAR analysis.

For an SGTR accident, the source of radioactivity for the release to the environment consists of primary coolant which is assumed to be at the technical specification values for dose equivalent ^{131}I when the SGTR accident occurs. Primary coolant is released through the ruptured tube to the secondary side of the steam generator. A portion of the leakage from the ruptured tube flashes and is released immediately to the environment with an assumed partition factor of 1. The remaining portion of the tube leakage which does not flash is mixed with the liquid in the affected steam generator and is released by the steaming process with a partition factor of 0.01. At the same time that the ruptured flow is being released to the affected steam generator, primary to secondary leakage is occurring at the technical specification allowed rate of 0.333 gpm to each intact steam generator. This radioactivity is also released to the environment via the steaming process and the release of iodine isotopes would also see a partition factor of 0.01. The licensee assumed that it would take 25 minutes to identify and isolate the faulted steam generator. As in the case of the MSLB accident, it was assumed that leakage would occur for 36 hours from the MSIVs above the seat drains.

The licensee's assessment of the consequences of an SGTR accident involved two cases. One was a pre-existing spike and the other was an accident-initiated spike. In both cases, it was assumed that primary to secondary leakage existed at the present technical specification value of 1 gpm total at operating conditions prior to and following the SGTR.

In the pre-existing iodine spike case it was assumed that the iodine spike occurred prior to the SGTR. The licensee performed its assessment of the consequences of this accident at the existing technical specification value for the maximum instantaneous value of dose equivalent ^{131}I in primary coolant, 60 µCi/gm. For the accident-initiated spike case, the licensee assumed that the SGTR initiates a concurrent iodine spike. For this assessment the licensee assumed that the reactor coolant was at the 100-hour technical specification limit for dose equivalent ^{131}I, 1 µCi/g when, concurrent with the SGTR, an iodine spike was assumed to occur that results in the release of iodine from the fuel to primary coolant. The release of iodine from the fuel was assumed to be at a rate that is 500 times the iodine release rate associated with the 100-hour technical specification value. In both the pre-existing and accident-initiated spike cases, the secondary system activity was assumed to be at the technical specification normal operation limit of 0.1 µCi/gm of dose equivalent ^{131}I. For both cases, the licensee assumed that it took 8 hours to cool the reactor down to a point where no further release of steam or radioactivity to the environment would occur. The licensee's analysis of the SGTR concluded that this accident resulted in no failed fuel.

For the accident-initiated spike the licensee halted the spike at 11,676 Ci of ^{131}I. The licensee stated as the basis for halting the spike when primary coolant reached this activity level was that Standard Review Plan Section 15.6.3 provides no guidance on the duration of the spike and that the typical approach is to terminate the spike when the primary coolant activity of dose equivalent ^{131}I is raised to the technical specification upper limit as defined for transients (60 µCi/g). The licensee also asserted that a majority of the thyroid dose from an SGTR is from the release of the flashed break flow directly to the atmosphere and from the release of radioactivity collected in the steam generator with the ruptured tube. Further, the licensee's analysis showed that break flow stopped flashing at 1970 seconds. After that time, break flow is mixed with the liquid in the steam generator and subject to partitioning prior to release to atmosphere. Break flow itself terminates at 5024 seconds and the spike was terminated at 5076 seconds. The contribution of the activity released via primary to secondary leakage to the intact steam generators is a very small contributor to the total dose. Therefore, extending the duration of the spike until residual heat removal (RHR) is initiated was determined to have little impact upon the dose consequences. The EAB dose is limited to the first 2 hours following the accident. The impact upon the LPZ and control room doses would be greater than upon the EAB dose since primary to secondary leakage is continuing to the intact steam generators for the duration of the accident and the activity of dose equivalent ^{131}I in primary coolant continues to increase. However, the licensee concluded that the increase in thyroid dose is still minimal. The licensee estimated the increases were 0.001 rem, 0.18 rem and 0.03 rem for the EAB, LPZ, and control room, respectively.

The staff has performed its assessment of the potential consequences of an SGTR event. The staff's assessment assumed that the reactor coolant activity level for dose equivalent ^{131}I was at 60 µCi/g for the pre-existing spike case and at 1 µCi/g for the accident-initiated spike case. The staff assumed that the partition factor for non-flashed releases from the intact and faulted steam generators was 0.01.

The staff agrees with the licensee's conclusion that the contribution to doses resulting from the intact steam generators is small. Phenomenologically speaking, there is no basis for stopping the iodine spike when the primary coolant activity level reaches 60 µCi/g of dose equivalent ^{131}I. It would make sense to halt the spiking if the assumed duration associated with the spiking resulted in the release of more iodine than is contained in the reactor core. However, the staff concluded that such a situation does not exist in this application. While the staff's independent assessment that ran the analysis until RHR was initiated determined that the doses are acceptable in this instance because the contribution from the intact steam generators is small and relative to the margin to the regulatory limits is negligible, the staff's position is that future licensee analyses should incorporate the iodine spiking factor until RHR is initiated.

In the staff's review, the licensee was asked whether it had assessed the impact of the reduced feedwater temperature on flashing fraction and break flow and the potential for steam generator overfill. The licensee responded that the overfill analysis for the replacement steam generator had included the impact of the reduced feedwater temperature. They indicated that the reduced feedwater temperature resulted in a small benefit with respect to offsite doses and, therefore, was not modeled in the analysis. The licensee also indicated that the scope of its submittal associated with the review of this USQ was the radiological impacts of operation of the Model E steam generators at reduced feedwater temperatures and the operation with the replacement steam generators at feedwater temperatures as low as 390 ^{o}F and that other impacts (i.e., flashing fraction, breakflow, and overfill) associated with these changes were being handled by 10 CFR 50.59 evaluations and submittals, as appropriate.

The staff also noted that the licensee's submittal had indicated that only 862 seconds were required to cool the plant down via the secondary side in the event of an SGTR. In response to an NRC question concerning this value, the licensee indicated that the 862 seconds is a code-calculated time. The 862 seconds is the time required to cool the reactor coolant system by dumping steam from the three intact steam generators to the temperature corresponding to saturation conditions at the pressure in the ruptured steam generator at the start of the cooldown and it incorporates subcooling margin and uncertainties associated with the emergency operating procedures. The licensee indicated that additional cooldown time is required and that the calculations accounted for the time to cool the plant to the point at which the RHR system may be used and steam releases from the secondary side end. Since the licensee's calculations accounted for the time to cool the plant to the point at which RHR is initiated, the staff had no further questions on the 862 seconds calculated by the code.

The staff performed confirmatory calculations of the radiological consequences of an SGTR accident. Table 2.1.2-1 presents the assumptions utilized by the staff in its assessment. Some of the information that is contained in Table 2.1.1-1 for the MSLB is appropriate for inclusion in the SGTR analysis. The potential consequences of an SGTR accident are presented in Tables 5-1 through 5-3. The staff concluded that, even with the reactor coolant activity levels at 60 µCi/g for the pre-existing spike case and 1 µCi/g for the accident-initiated spike case, doses were found to be acceptable. The staff's calculations show that the thyroid doses for the EAB and LPZ are within the guidelines in the SRP (see page 2) for this accident, and the doses remain below the 10 CFR Part 100 dose criteria for offsite consequences. Also, the staff's calculated doses to control room personnel were below the GDC 19 criteria for control room doses. Therefore, the staff concluded that the radiological consequences associated with the SGTR were acceptable.

2.1.3 Locked Rotor

The licensee postulated the seizure of a reactor coolant pump shaft. Such a seizure could occur from either a mechanical failure or the loss of component cooling flow to the pump shaft seals. This event is most limiting when it occurs at full power. For this case, flow would be asymmetrically reduced. With the reduction in pump flow, core temperature increases. Assuming a positive moderator temperature coefficient, core power will increase. This results in a slight decrease in core average heat flux. Insertion of the control rod element assemblies due to a low reactor coolant system trip terminates the power rise but will result in some fuel pins experiencing DNB for a short period. These pins will experience fuel failure. The gap activity associated with these failed fuel pins will be released to primary coolant. The licensee assumed that 10 percent of the fuel experienced DNB. However, they also performed a radiological analysis which assumed that 15 percent of the fuel experienced DNB.

In the licensee's assessment it was assumed that the sources of radioactivity that are released to the environment include the secondary side activity contained in the steam generators and the activity associated with primary to secondary leakage to these steam generators. For the activity associated with primary side to the secondary side leakage, primary coolant activity was assumed to be at the technical specification value for dose equivalent ^{131}I. However, the contribution of activity from the initial reactor coolant activity levels, even when it is assumed that the reactor coolant activity levels are at the technical specification values for dose equivalent ^{131}I and an iodine spike is considered, is small relative to the contribution of activity from the release of the gap inventory from either 10 or 15 percent of the fuel rods in the core experiencing DNB.

The licensee's analysis assumed that offsite power was lost and that the main condensers were unavailable for steam dump operation. Primary to secondary leakage was assumed to occur for 8 hours following the accident. At 8 hours after the accident, the RHR system would start to cool the plant and no steam or activity was assumed to be released to the environment.

The licensee indicated that dose analyses for the Model E steam generators at the 420 ^{o}F feedwater temperature and the Model Δ94 steam generators showed that the thyroid doses were about 1/3 lower than the current UFSAR analysis but that the whole-body and skin doses for the LPZ are higher than the current analysis. The bases for these differences were the following:

1. The new analyses use a lower steam release rate and a larger steam generator mass than the existing analysis which results in a dose reduction of approximately 1/3.

2. The existing analysis utilized a dose model that depleted the radioactivity in the steam generator during the steam releases while the new analyses do not. This resulted in EAB doses that were higher by 1/3 at the EAB and a factor of 2 higher at the LPZ.

3. The existing analysis was performed in two parts. The doses due to iodine were calculated and then the doses due to the noble gases were calculated. The new analyses were performed with the contributions from all radionuclides determined.

   The licensee indicated that at the EAB the 1/3 decrease in steam release rate was offset by the use of the non-depleting model rather than the depleting model. Therefore, the licensee anticipated the skin and whole body to be about the same for both analyses. However, for the LPZ, the non-depleting model has a dominating effect because the time frame has been increased from 2 to 8 hours. Although the steam flow is reduced by a factor of 1/3, the non-depleting model increases the doses by a factor of 2. Therefore, for the new analyses, the licensee assumed that the doses at the LPZ would be increased by 1/3 {(1-1/3) * 2} = 4/3 over the previous analyses.

4. For all analyses, an iodine partition factor of 0.01 was assumed for iodine released from the steam generator inventory via the steaming process. In the existing analysis, this reduction in iodine inventory was simulated by assigning a steam release rate that was 1 percent of the actual value. Because of the reduction in the steaming rate, the licensee indicated that there was no significant depletion in the activity of in the steam generator due to the release of steam.

   Because of the differences in steam release masses and steam generator mass, as noted in Item 1 above, there are differences in thyroid doses between the existing analysis and the revised analyses. The licensee's calculations showed that the thyroid dose from the bounding analysis for the Model E steam generators at 390 ^{o}F feedwater temperature) and the Model Δ94 steam generators (at 420 ^{o}F feedwater temperature) are about 1/3 lower than the Model E steam generators at 440 ^{o}F feedwater temperature).

The staff performed confirmatory calculations of the radiological consequences of the locked rotor accident. Table 2.1.3-1 presents the assumptions utilized by the staff in its assessment. The staff's assessment of the potential dose consequences of a locked rotor accident are presented in Tables 5-1 through 5-3. The staff's calculations show that the thyroid doses for the EAB and LPZ are within the guidelines in the SRP (see page 2) for this accident, and the doses remain below the 10 CFR Part 100 dose criteria for offsite consequences. Also, the staff's calculated doses to control room personnel were below the GDC 19 criteria for control room doses. Therefore, the staff concluded that the radiological consequences associated with the locked rotor accident were acceptable.

2.1.4 Rod Ejection

The licensee performed an assessment that assumed two release paths associated with a rod ejection accident. One pathway was assumed to occur via containment leakage. The second pathway was assumed to occur via leakage of primary coolant to the secondary side with the release to the environment from the secondary side. The licensee concluded that for the present analysis in the UFSAR and the two new analyses, the reduced feedwater temperature for the existing Model E steam generators of 420 ^{o}F and the Model Δ94 steam generators with a feedwater temperature of 390 ^{o}F, there would be no change in the whole-body and skin doses for the containment leakage pathway. However, the licensee concluded that there would be an increase in the whole-body and skin doses for the secondary side leakage pathway. This increase in whole-body and skin dose was caused by an increase in time required to equalize the pressure between the primary and secondary side (from 1250 seconds to 4500 seconds) and the decrease in time required to release the contents of the steam generator (300 seconds to 191 seconds). The licensee's revised analysis assumes that the minimum time to release the initial steam generator mass is 300 seconds that results in an average release rate of 14,460 lbs/min. However, the time needed to equalize the pressure between the primary and secondary sides is 1250 seconds. Therefore, the primary to secondary leak rate of 1 gpm will continue out to 1250 seconds. The licensee performed its analysis assuming that the release from the steam generators would continue out to 1250 seconds at a rate of 14,460 lbs/min.

In the licensee's development of the bounding analysis, its internal review resulted in questions on the source document for the steam releases that was contained in the original analyses. Consequently, the licensee decided, based upon the low radiological consequences associated with the rod ejection accident, to assume a conservative set of values for the steam releases. The licensee assumed that the entire contents of the steam generators were released. The minimum time for such a release to occur was 191 seconds. This results in an average steam generator release rate of 207,000 lbs/min. However, the time to equalize pressure between the primary and secondary sides is increased to 4500 seconds. Therefore, the licensee assumed that the primary to secondary leak would continue for 4500 seconds at a rate of 207,000 lbs/min. The licensee considered this assumption to be overly conservative since the steaming rate will decrease significantly over time.

The licensee concluded that the release from the secondary side had a small effect on the thyroid doses. The major impact was on the whole-body and skin doses with the large steam release accounting for this increase.

In response to the staff's April 14, 1999, request for additional information, the licensee determined that the existing analysis in the UFSAR contained an error in the determination of the amount of ^{131}I in the fuel pin gap. The licensee had assumed that the fraction in the gap was 0.12, which is the value for extended burnup fuel for a fuel handling accident in NUREG/CR-5009 ("Assessment of the Use of Extended Burnup Fuel in Light-Water Power Reactors"). However, the licensee had concluded, based upon its review of NUREG/CR-5009 that this assumption was inappropriate for an accident involving core damage. Consequently, the licensee concluded that the appropriate value was 0.10.

The staff evaluated the consequences of a rod ejection accident. The staff concluded that the licensee’s assumption of 0.10 for the ^{131}I may be appropriate for an accident involving the clad damage associated with a rod ejection accident, but the licensee did not provide sufficient justification to allow the staff to draw such a conclusion. This issue is currently being reviewed by both the NRC and the industry to refine the gap fraction that should be used in conducting this type of analysis. When this issue is resolved and agreed to by both the NRC and the industry, the licensee should evaluate the impact that resolution has on its analysis. However, the currently accepted NRC position is that a gap fraction of 0.12 should be used. Using a gap fraction of 0.12 provides a higher dose consequence than the 0.10 used in the licensee's analysis. Consequently, the staff’s assessment incorporated a gap fraction of 0.12. While the staff's independent assessment determined the doses are acceptable in this instance because the margin between the doses calculated by the staff to the regulatory limits are substantial, the staff's position is that future licensee analyses should use a iodine gap fraction of 0.12, provide additional justification for using 0.10, or use the gap fraction, with supporting justification, resulting from the review of this issue by the NRC and the industry.

The staff’s assessment included the consequences associated with releases occurring from containment leakage and secondary side release. Table 2.1.4-1 presents the assumptions utilized by the staff in its assessment. The potential dose consequences of a rod ejection accident are presented in Tables 5-1 through 5-3. The staff's calculations show that the thyroid doses for the EAB and LPZ are within the guidelines in the SRP (see page 2) for this accident, and the doses remain below the 10 CFR Part 100 dose criteria for offsite consequences. Also, the staff's calculated doses to control room personnel were below the GDC 19 criteria for control room doses. Therefore, the staff concluded that the radiological consequences associated with the rod ejection accident were acceptable.

2.1.5 Miscellaneous Analyses

The licensee assessed the consequences of the feedwater temperature reduction on other accidents such as the LOCA, waste gas decay tank release, the feedwater system pipe break, reactor coolant pump shaft break, small line break outside containment, and liquid radwaste tank processing system failure and on primary side and secondary side isotopic quantities. The licensee concluded that the impact of the reduced feedwater temperature would be negligible in each of these areas. The staff reviewed the licensee's analysis and concluded that the licensee correctly found that reduced feedwater temperature had a negligible effect on the radiological consequences of these accidents.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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. 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 64124). 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.

5.0 CONCLUSION

The staff has assessed those accidents that were identified as having USQs as a consequence of the decrease in feedwater temperature. The dose consequences are presented in Tables 5-1 through 5-3. The results of the staff's assessment demonstrated that, for those accidents that are impacted by the reduction in feedwater operating temperature, the doses would not exceed the dose guidelines presently contained in the Standard Review Plan, 10 CFR Part 100, or GDC 19 of 10 CFR Part 50, Appendix A, for either offsite locations or control room operators. The staff concluded that the revision to the offsite dose licensing bases to account for operation of the existing steam generators at reduced feedwater inlet temperatures and operation with the new replacement steam generators, also at a reduced feedwater temperature, are acceptable and may be incorporated into the UFSAR.

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.

Principal Contributor: John J. Hayes

Date: March 2, 2000

Attachments:
1. Table 2.1.1-1 Staff's Input Parameters for STP MSLB Accident
2. Table 2.1.2-1 Staff's Assumptions for STP SGTR Accident
3. Table 2.1.3-1 Staff's Assumptions for STP Rod Ejection Accident
4. Table 2.1.4-1 Staff's Assumptions for STP Locked Rotor Accident
5. Table 5-1 Thyroid Doses for STP from Postulated Accidents
   Table 5-2 Whole Body Doses for STP from Postulated Accidents
6. Table 5-3 Control Room Operator Doses for STP from Postulated Accidents

TABLE 2.1.1-1
Staff's Input Parameters for South Texas Project Main Steam Line Break Accident

1. Primary coolant isotopic activity @ 60 µCi/g of dose equivalent ^{131}I.

   Pre-existing Spike Value (µCi/g)

   ^{131}I = 46.3
   ^{132}I = 54.0
   ^{133}I = 73.3
   ^{134}I = 11.0
   ^{135}I = 40.5

2. Mass of primary coolant and secondary coolant (lbs) 585,978

3. Technical Specification limits for DOSE EQUIVALENT ^{131}I in the primary and secondary coolant.

   Primary Coolant DOSE EQUIVALENT ^{131}I concentration (µCi/g)
   Maximum Instantaneous Value 60
   100 hour Value 1.0

   Secondary Coolant DOSE EQUIVALENT ^{131}I concentration (µCi/g) 0.1

4. Technical Specification for the primary to secondary leak rate.

   Primary to secondary leak rate, faulted STEAM GENERATOR (gpm) 0.35
   Primary to secondary leak rate, intact steam generators (gpm) 0.65

5. Iodine Partition Factor

   Affected Steam Generator 1
   Intact Steam Generators 0.01

6. Secondary Side Steam Released to the Environment

   0 - 2 hours (lbs) 4.52E5
   2 - 8 hours (lbs) 1.08E6

7. MSIV Leakage Above Seat Drains on Intact steam generators (lb_m/min) 347.4

8. Letdown Flow Rate (gpm) 100

9. Equilibrium Release Rate from Fuel for 1 µCi/g of Dose Equivalent ^{131}I

   Ci/hr
   ^{131}I = 18.4
   ^{132}I = 93.0
   ^{133}I = 38.7
   ^{134}I = 42.6
   ^{135}I = 33.9

TABLE 2.1.1-1--Continued

10.	Atmospheric Dispersion Factors	
	EAB (0-2 hours)	1.4E-4
	LPZ (0-8 hours)	1.9E-5
11.	Spiking Factor for Accident Initiated Spike	500
12.	Control Room Parameters	
	Filter Efficiency (percent)	
	Intake Adsorber	99
	Intake HEPA	99
	Recirculation Adsorber	95
	Recirculation HEPA	99
	Volume (ft^3)	274,080
	Makeup flow (cfm)	1800
	Recirculation Flow (cfm)	9,000±10 percent
	Unfiltered Inleakage (cfm)	10
	Occupancy Factors	
	0-1 day	1.0
	1-4 days	0.6
	Atmospheric Dispersion Factors (sec/m^3)	1.7E-2

Table 2.1.2-1
Staff's Assumptions for South Texas Project Steam Generator Tube Rupture Accident

Parameter	Value
Iodine Partition Factor	0.01
Steam Release from Affected Steam Generator	
0-2 hours (lbs)	213,000
2-8 hours (lbs)	35,200
Steam Release from Intact Steam Generator (lbs)	
0-2 hours	633,300
2-8 hours	1,322,600
Mass of Break Flow Released to Affected Steam Generator (lbs)	
0 - 5,020 seconds	136,100
Primary Coolant Activity Level - Dose Equivalent ^{131}I (µCi/g)	
Pre-existing Spike	60
Accident Initiated Spike	1
Flashing Fraction 0 - 5,020 seconds	Variable, refer to Attachment (1) of May 14, 1999, South Texas Project Letter
Time to Isolate Faulted Steam Generator (min)	25
MSIV Leak Rate above Seat Drains (lb_m/min)	
Intact Steam Generators	
0 - 36 hours	347.4 for all 3 steam generators
Faulted Steam Generator	
0 - 36 hours	115.8

Table 2.1.3-1
Staff's Assumptions for South Texas Project Rod Ejection Accident

Parameter	Value
Core Thermal Power (MWt)	4100
Primary to Secondary Leak Rate (gpm)	1
Clad Failure (percent of core fuel)	10
Activity release to reactor coolant from failed fuel and available for release (percent of gap inventory)	100
Fraction of fuel activity in the gap	
^{131}I	0.12
^{85}Kr	0.3
Others	0.1
Melted Fuel (percent of core)	0.25
Activity released to reactor coolant from melted fuel and available for release (percent of core inventory)	0.25 for noble gases 0.0625 for iodine
Iodine Partition Factor in the SGs before and after the accident	0.01
Containment Volume (ft3)	1.84E6
Containment Leak Rate (percent/day)	0.3 for t = 0-1 day 0.15 for t > 1 day
Iodine Form in Containment (fraction)	
Particulate	0.05
Organic	0.04
Elemental	0.91
Time between Accident and Equalization of Primary to Secondary System Pressure (sec)	4500
Primary Coolant Mass (lbs)	585,978
Secondary Side Steam Release (lbs)	1,560,000
Minimum Secondary Steam Flow Rate (lbs/hr)	15,740,000
Secondary Side Mass (lbs)	659,412

Table 2.1.4-1
Staff's Assumptions for South Texas Project Locked Rotor Accident

Parameter	Value
Core Thermal Power Level (MWt)	4100
Duration of Plant Cooldown by Secondary System (hr)	8
Gap Fraction:	
^{131}I	0.12
^{85}Kr	0.30
All others	0.10
Failed Fuel Rods (percent)	15
Primary to Secondary Leak Rate @ Operating Conditions (gpm)	1
Iodine Partition Factor in steam generators	0.01
Steam Released from steam generators (lbs)	
0-2 hours	455,047
2-8 hours	1,137,757
Steam Flow (lbs/hr)	15,740,000
Primary Coolant Mass (lbs)	585,978
Total of Steam Generators Mass (lbs)	659,412

Table 5-1
Thyroid Doses for South Texas Project from Postulated Accidents (Rem)

	Accident	EAB	LPZ
1.	Main Steam Line Break		
	Coincident Spike	0.14	0.032
	Pre-existing Spike	0.042	0.03
2.	Steam Generator Tube Rupture		
	Coincident Spike	3.5	0.58
	Pre-existing Spike	12	1.9
3.	Locked Rotor	6.8	0.92
4.	Rod Ejection		
	Secondary Side	11	1.6
	Containment Leakage	24	30

Table 5-2
Whole Body Doses for South Texas Project from Postulated Accidents (Rem)

	Accident	EAB	LPZ
1.	Main Steam Line Break		
	Coincident Spike	<1	<1
	Pre-existing Spike	<1	<1
2.	Steam Generator Tube Rupture		
	Coincident Spike	<1	<1
	Pre-existing Spike	<1	<1
3.	Locked Rotor	<1	<1
4.	Rod Ejection		
	Secondary Side	<1	<1
	Containment Leakage	<1	<1

Table 5-3
Control Room Operator Doses for South Texas Project from Postulated Accidents (Rem)

	Accident	Thyroid	Whole Body
1.	Main Steam Line Break		
	Coincident Spike	0.14	1.5
	Pre-existing Spike	0.14	1.4
2.	Steam Generator Tube Rupture		
	Coincident Spike	2.6	<1
	Pre-existing Spike	8.10	<1
3.	Locked Rotor	3.9	1.2
4.	Rod Ejection		
	Secondary Side	6.6	<1
	Containment Leakage	6.9	<1