February 20, 2007

Mr. M. R. Blevins
Senior Vice President
& Chief Nuclear Officer
TXU Power
ATTN: Regulatory Affairs
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: NEW METHODS AND ASSUMPTIONS FOR RADIOLOGICAL CONSEQUENCES CALCULATIONS (TAC NOS. MC8163 AND MC8164)

Dear Mr. Blevins:

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. NPF-87 and Amendment No. 130 to Facility Operating License No. NPF-89 for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, respectively. The amendments consist of changes to the Final Safety Analysis Report (FSAR) in response to your application dated August 22, 2005, as supplemented by letters dated September 18 and October 23, 2006.

The amendments revise the FSAR Sections 1, 6, and 15. The changes reflect the licensee's adoption of Nuclear Regulatory Commission's (NRC's) Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," for calculating radiological consequences, and replacement of CPSES, Unit 1, steam generators in the spring of 2007. The amendments would supercede the TXU Generation Company LP's corresponding radiological consequence analyses based on other NRC regulatory guides.

Based on its review, the NRC staff finds that the methods of analysis and assumptions used by the licensee for radiological dose consequence analyses are consistent with the NRC staff's conservative requirements and regulatory guidance, and are acceptable for implementation in the CPSES's licensing basis.

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/

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 130 to NPF-87
2. Amendment No. 130 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
License No. NPF-87

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

   A. The application for amendment by TXU Generation Company LP dated August 22, 2005, as supplemented by letters dated September 18 and October 23, 2006, 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.

2. Accordingly, Amendment No. 130 revises the Final Safety Analysis Report Sections 1, 6, and 15 to reflect the licensee’s adoption of Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” and replacement of Comanche Peak Steam Electric Station, Unit 1, steam generators in the spring of 2007; as outlined in the application dated August 22, 2005, and as supplemented by letters dated September 18 and October 23, 2006. The Amendment supercedes the licensee’s corresponding radiological analysis assumptions based on other regulatory guides.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License

Date of Issuance: February 20, 2007

TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
License No. NPF-89

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

   A. The application for amendment by TXU Generation Company LP dated August 22, 2005, as supplemented by letters dated September 18 and October 23, 2006, 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.

2. Accordingly, Amendment No. 130 revises the Final Safety Analysis Report Sections 1, 6, and 15 to reflect the licensee's adoption of Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," and replacement of Comanche Peak Steam Electric Station, Unit 1, steam generators in the spring of 2007; as outlined in the application dated August 22, 2005, and as supplemented by letters dated September 18 and October 23, 2006. The Amendment supercedes the licensee's corresponding radiological analysis assumptions based on other regulatory guides.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License

Date of Issuance: February 20, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 130

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 130

TO FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Facility Operating Licenses, Nos. NPF-87 and NPF-89, with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Facility Operating License No. NPF-87

REMOVE	INSERT
- 3 -	- 3 -

Facility Operating License No. NPF-89

REMOVE	INSERT
- 3 -	- 3 -

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 130 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated August 22, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052380403), as supplemented by letters dated September 18 (ADAMS Accession No. ML062700200) and October 23, 2006 (ADAMS Accession No. ML063040564), TXU Generation Company LP (TXU, the licensee) requested changes to the Final Safety Analysis Report (FSAR) for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The supplements dated September 18 and October 23, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The proposed changes would revise the FSAR Sections 1, 6, and 15. These changes support the licensee's adoption of Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors", and replacement of CPSES Unit 1 steam generators in the spring of 2007. The licensee did not propose to make any changes to the CPSES Units 1 and 2 technical specifications with this submittal, as noted by the licensee's supplementary letter dated October 23, 2006.

Adoption of RG 1.195 is contingent upon adoption of RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors." The adoption of RG 1.196 and Technical Specification Task Force (TSTF)-448, Revision 3, "Control Room Habitability," will be implemented through approval of TXU License Amendment Request (LAR) 06-011, which was submitted on October 23, 2006.

2.0 REGULATORY EVALUATION

This safety evaluation input addresses the impact of the proposed changes on previously analyzed design-basis accident (DBA) radiological consequences, and the acceptability of the revised analysis results. The regulatory requirements on which the Nuclear Regulatory

Commission (NRC) based its acceptance are the accident dose guidelines of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 100.11 and 10 CFR Part 50 Appendix A, General Design Criterion 19 (GDC 19), "Control room." The NRC staff also considered relevant information in the CPSES FSAR.

The regulatory framework for requesting this licensing action is based on RG 1.195. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.195. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

- Section 100.11 of 10 CFR, "Determination of exclusion area, low population zone, and population center distance."
- Part 50, Appendix A of 10 CFR, "General Design Criteria for Nuclear Power Plants," GDC 19, "Control room."
- RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors."
- Standard Review Plan (SRP) Section 6.4, "Control Room Habitability System."
- SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System."
- SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Appendix A.
- SRP Section 15.3.3 -15.3.4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break."
- SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Appendix A.
- SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment."
- SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Rupture (PWR)."
- SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Appendix A and Appendix B.
- SRP Section 15.7.3, "Postulated Radioactive Release Due to Liquid-Containing Tank Failures."
- SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents."
- SRP 11.3, "Gaseous Waste Management Systems."

The NRC staff also considered relevant information in the CPSES FSAR and technical specifications.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design-Basis Accidents

TXU analyzed the following DBAs for the radiological consequences in the main control room (MCR) at the emergency area boundary (EAB) and at the outer boundary of the low-population zone (LPZ):

1. Loss-of-Coolant Accident (LOCA)
2. Fuel Handling Accident (FHA)
3. Steam Generator Tube Rupture (SGTR)
4. Main Steam Line Break (MSLB)
5. Primary Coolant Pump Locked Rotor Accident (LRA)
6. Control Rod Ejection Accident (CREA)
7. Chemical and Volume Control System (CVCS) Letdown Line Break
8. Gas Decay Tank Rupture
9. Liquid Waste Tank Rupture

The licensee performed calculations to determine the thyroid and whole body doses at the EAB for the worst 2-hour period following the onset of the accident. The integrated doses at the outer boundary of the LPZ and the integrated doses in the CPSES MCR were evaluated for the duration of the accident. TXU performed all the radiological consequence calculations for the LAR with the RADTRAD 3.03 computer code.

The NRC staff performed independent confirmatory dose calculations for events 1 through 6 using the NRC-sponsored radiological consequence computer code, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors.

The NRC staff performed an audit of the licensee's calculations focusing on the methodology, current licensing basis, and assumptions for events 1 through 9.

3.1.1 Loss-of-Coolant Accident

LOCA Dose Results						
Dose Roentgen Equivalent Man (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
Whole Body (WB)	0.79	25	0.29	25	2.3	5
Thyroid	63.0	300	47.0	300	43	50
Beta					16.0	50

The radiological consequence design-basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary reactor coolant system (RCS) piping. The accident scenario assumes the deterministic failure of the emergency core cooling system (ECCS) to provide adequate core cooling, which results in a significant amount of core damage. This general scenario does not represent any specific accident sequence, but is representative of a class of severe core damage incidents. Such a scenario would be expected to require multiple failures of systems and equipment, and lies beyond the severity of incidents evaluated for design-basis transient analyses.

In the evaluation of the LOCA design-basis radiological analysis, the licensee included dose contributions from the following sources:

- Containment leakage
- Engineered safety feature (ESF) system component leakage

The licensee's analysis resulting in thyroid and whole body doses for each of the above release pathways was added together to determine the total projected thyroid and whole body doses for the LOCA. The licensee also calculated the beta radiation dose in the control room (CR) in a similar manner.

During a design-basis LOCA, it is conservatively assumed that the fission product release to the containment will occur at the start of the accident and is likely to mix instantaneously and homogeneously throughout the free air volume of the primary containment.

3.1.1.1 Containment Sprays

The TXU design-basis LOCA analysis credits the use of containment sprays to remove elemental and particulate iodine from the containment atmosphere. To model the spray removal, the licensee assumed that the CPSES containment is separated into five regions. Each containment region is then divided into sprayed and unsprayed volumes. The analysis assumes that the total sprayed volume of all five regions of the primary containment equals 56.3 percent of the total containment volume, in accordance with the current licensing basis for

CPSES. The mixing rate between the total sprayed and total unsprayed volumes of containment during spray operation is assumed to be two turnovers of the total unsprayed volume per hour, which is consistent with the guidance of RG 1.195.

The elemental and particulate removal coefficients have been revised. The licensee did not provide the NRC staff with the calculations it performed to determine the revised removal coefficients. The current licensing basis elemental iodine removal coefficient values for the four sprayed regions of the primary containment are much greater than 10 hr^{-1}. These values have increased with this submittal, but the licensee has limited the elemental iodine removal coefficient to a maximum value of 10 hr^{-1}. This is consistent with the guidance of RG 1.195.

The submitted CPSES FSAR markup contains the new value for the total collection efficiency for a single drop (E) divided by the drop diameter (d) for the particulate iodine removal coefficient. The proposed E/d value of 10/meter (m) is used until airborne activity is reduced by a factor of 50 and then a value of 1.0/m is used. This is consistent with SRP Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” for particulate iodine removal. The lowest calculated particulate iodine removal coefficient for any sprayed region of the primary containment is 11.4 hr^{-1}. After the particulate iodine inventory has been reduced by a factor of 50 a value of 1.14 hr^{-1} is applied. These values are conservatively applied to the entire sprayed volume of the containment. The methodology used to calculate particulate iodine removal remains unchanged from the current licensing basis, and is consistent with the SRP Section 6.5.2.

3.1.1.2 Containment Leakage

The total containment leakage for CPSES is 0.10 weight percent per day, as governed by technical specifications (TS). The containment maximum allowable leakage rate (L_a) is reduced by one-half to 0.05 weight percent per day at 24 hours for the duration of the accident, which is consistent with RG 1.195. The containment leakage is modeled as a ground level release to the environment.

3.1.1.3 Engineered Safety Feature Leakage

During a LOCA, a portion of the fission products released from the fuel will be carried to the containment sump via spillage from the RCS, by transport of activity from the containment atmosphere to the sump by containment sprays. During the initial phases of a LOCA, safety injection (SI) and the containment spray systems draw water from the refueling water storage tank (RWST). Several minutes after accident initiation valve realignment occurs to switch the suction water source for the ESF systems from the RWST to the containment sump. This recirculation flow causes contaminated water to be circulated through piping and components outside of the containment, where small amounts of system leakage could provide a path for the release of fission products to the environment. The licensee’s current licensing basis uses a value of 2 gallons per minute (gpm) for the evaluation of the ESF leakage contribution to the LOCA dose.

To evaluate the radiological consequences of ESF leakage, the licensee used the deterministic approach as prescribed in RG 1.195. This approach assumes that 50 percent of the iodine originally present in the core is released from the fuel to mix instantaneously and homogeneously in the containment sump water.

The licensee assumed that the leakage of recirculating sump fluids commences at 10 minutes, which is the earliest time that the recirculation of contaminated fluids would begin. The licensee conservatively used a flashing fraction of 0.1 for the ESF leakage calculation for the duration of the event. As a result, 10 percent of the entrained iodine activity in the ESF leakage effluent is assumed to be released to rooms housing the leaking components and then immediately swept away by the ventilation system and released to the atmosphere. In accordance with RG 1.195, the licensee assumed that the chemical form of the released iodine is 97-percent elemental and 3-percent organic.

3.1.1.4 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this safety evaluation (SE), respectively.

3.1.1.5 LOCA Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and GDC 19. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in RG 1.195. The assumptions found acceptable to the NRC staff are presented in Table 4 and in Table 1 (both tables are at the end of this SE). The NRC staff performed independent confirmatory dose evaluations and an audit of the licensee's analyses to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the LOCA meet the applicable accident dose acceptance criteria in RG 1.195, as noted in Table 1, and are, therefore, acceptable.

3.1.2 <u>Fuel Handling Accident</u>

FHA Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.087	6.3	0.013	6.3	0.13	5
Thyroid	22	75	3.3	75	3.6	50
Beta					4.1	50

The FHA analysis postulates that a spent fuel assembly is dropped during fuel handling and all of the fuel rods in the dropped assembly are conservatively assumed to experience fuel cladding damage, releasing the radionuclides within the fuel rod gap to the fuel pool or reactor cavity water. The affected assemblies are assumed to be those with the highest inventory of fission products in the core. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP), depending on their physical and chemical form.

The licensee conservatively assumed no decontamination for noble gases and retention of all aerosol and particulate fission products in the overlying water. As prescribed in RG 1.195, the FHA is analyzed based on the assumption that 100 percent of the fission products released from the reactor cavity or SFP are released to the environment in 2 hours. The licensee did not credit filtration, holdup, or dilution of the released activity. Since the revised assumptions and inputs are identical for the FHA within containment and the FHA outside containment, the FHA outside containment is considered bounding.

3.1.2.1 SFP Decontamination Factor (DF)

TXU made changes to their current licensing basis assumptions for the SFP DF, maximum fuel rod internal pressure, and depth of water above the fuel in the SFP. The SFP DF increased from 100 to 160, but is less than the RG 1.195 assumed DF of 200. The maximum fuel rod internal pressure increased from 1200 pounds per square inch gauge (psig) to 1500 psig. The depth of water above the fuel in the SFP decreased from 23 feet (ft) to 21 ft.

RG 1.195 Appendix B gives a pool effective iodine decontamination factor of 200 as an acceptable assumption for the FHA, with a restriction that this value is acceptable for water depths of at least 23 ft and fuel internal rod pressures up to 1200 psig. If the depth of water is not at least 23 ft, the DF will have to be determined and found acceptable to the NRC staff on a case-by-case basis. Additionally, RG 1.195 allows the calculation of SFP DF on a case-by-case basis for rod internal pressures of greater than 1200 psig. The calculations supporting the new SFP DF of 160 were not included in this LAR submittal but were provided for the NRC staff to audit.

The licensee used the methodology specified in Westinghouse Topical Report WCAP-7828, “Radiological Consequences of a Fuel Handling Accident,” dated December 1971, and CN-CRA-98-063, “Generic Determination of Impact of Increased Fuel Rod Pressure on Pool Scrubbing Removal of Iodine for Fuel Handling Accident,” dated July 1998. TXU calculated a DF of 278, which was then conservatively rounded to 270. From this an overall DF of 161.4 was calculated which was then again conservatively rounded to 160.

SRP 15.7.4 states that if factors less conservative than those recommended by RG 1.25 are used, guidance provided by G. Burley, Radiological Safety Branch, Division of Reactor Licensing, NRC, titled, “Evaluation of Fission Product Release and Transport for a Fuel Handling Accident,” revised October 5, 1971, should be consulted to determine if an adequate basis for the proposed deviation exists. This evaluation is based, in part, on an earlier Westinghouse Topical Report WCAP-7518, “Radiological Consequences of a Fuel Handling Accident,” dated June 1970. The basis of RG 1.25 and ultimately RG 1.195 utilizes the experimental data and formulation of WCAP-7518 in a manner to ensure a conservative result appropriate for licensing purposes. The methodology used in WCAP-7518 is similar to that of WCAP-7828 used by the licensee. The NRC staff found the assumptions and methodology used to calculate the SFP DF acceptable and thereby finds the new SFP DF value to be acceptable.

3.1.2.2 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.2.3 FHA Conclusion

The licensee evaluated the radiological consequences resulting from the postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and GDC 19 and the accident-specific dose acceptance criteria in RG 1.195. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in RG 1.195. The assumptions found acceptable to the NRC staff are presented in Tables 5 and 1 (both tables are at the end of this SE). The NRC staff performed independent confirmatory dose evaluations and audited the licensee's analyses to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable accident dose acceptance criteria in RG 1.195, as noted in Table 1, and are, therefore, acceptable.

3.1.3 <u>Steam Generator Tube Rupture</u>

SGTR Pre-Accident Iodine Spike Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.08	25	0.013	25	0.07	5
Thyroid	22.0	300	3.4	300	18	50
Beta					1.9	50
SGTR Concurrent Iodine Spike Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.077	2.5	0.012	2.5	0.069	5
Thyroid	8.8	30	1.4	30	2.0	50
Beta					1.9	50

In an SGTR accident, it is assumed that there is a complete severance of a single steam generator (SG) tube. The accident is assumed to take place at full power with the reactor coolant fission product concentrations corresponding to continuous operation with a limited amount of fuel damage. The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG (denoted by the licensee as the affected SG) with an assumed release to the environment through the SG Atmospheric Relief Valves (ARVs). For this accident scenario, a loss of off site power (LOOP) is assumed to occur concurrently with the tube rupture. Because the LOOP renders the main condenser unavailable, the plant is cooled down by release of steam to the environment. In the TXU analysis, the ARV on the affected SG is assumed to open to control SG pressure at the beginning of the event. After operator action is credited to close the affected SG ARV, the same ARV is assumed to fail fully open. The affected SG discharges steam to the environment for 30 minutes until the generator is manually isolated a second time by closure of the SG atmospheric dump block valve. Break flow into the affected SG continues until 30 minutes, at which time the RCS is at a lower pressure than the secondary system. Depressurization of the SG is necessary to allow residual heat removal system (RHRS) cooling.

The licensee evaluated the dose consequences from discharges of steam from the three intact SGs for a period of 8 hours, until the primary system has cooled sufficiently to allow an alignment to the RHRS. At this point in the accident sequence, steaming is no longer required for cooldown and releases from the intact SGs are terminated.

Appendix E of RG 1.195 identifies acceptable radiological analysis assumptions for an SGTR accident. If a licensee demonstrates that no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the TS. Two radioiodine spiking cases are considered. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For CPSES, the maximum iodine concentration allowed by the TS as a result of an iodine spike is 60 μCi/gm dose equivalent (DE) I-131.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. For CPSES, the RCS TS limit for normal operation is 0.45 μCi/gm DE I-131. The licensee used a value of 1.0 μCi/gm DE I-131 in their calculations, which gives bounding doses.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an SGTR accident. Therefore, consistent with the current licensing analysis basis and regulatory guidance, the licensee performed the SGTR accident analyses for the pre-accident iodine spike case and the concurrent iodine spike case.

3.1.3.1 Releases from the Affected SG

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the affected SG by the break flow. In the TXU analysis for CPSES, break flow is terminated after 30 minutes. A portion of the break flow is assumed to flash to steam because of the higher enthalpy in the RCS. The noble gas and iodine in the flashed portion of the break flow will ascend to the steam space of the affected generator and be available for release with no credit taken for scrubbing by the SG liquid. The radionuclides entering the steam space as the result of flashing past directly to the environment through the SG ARVs. The iodine and other non-noble gas isotopes in the non-flashed portion of the break flow are assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable partitioning factor. In accordance with the guidance from RG 1.195, the licensee's evaluation of the releases from the steaming of the liquid mass in the SG credits a partitioning factor of 0.01 for all non-noble gas isotopes. Following the applicable regulatory guidance, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation.

3.1.3.2 Releases from the Intact SGs

The licensee assumed that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the intact SGs by the leak rate limiting condition for operation (1 gpm) specified in the TS. All radionuclides in the primary coolant leaking into the intact SGs are assumed to enter the SG liquid. Radionuclides initially in the SG liquid, and those entering the SG liquid from the leakage flow, are released as a result of secondary liquid boiling/steaming, with a partitioning factor of 0.01 for all non-noble gas isotopes. Therefore, 1 percent of the iodines and particulates are assumed to pass into the steam space and then directly to the environment. The licensee assumed that all noble gases that are released from the primary system to the intact SGs are released to the environment without reduction or mitigation. Releases were assumed to continue from the intact SGs for a period of 8 hours until the RHRS is placed in service. The 8-hour steaming period is based on the time necessary to cooldown crediting safety grade equipment only.

3.1.3.3 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.3.4 SGTR Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and GDC 19 and the accident-specific dose acceptance criteria in RG 1.195. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in RG 1.195. The assumptions found acceptable to the NRC staff are presented in Table 6 and Table 1 (both tables are at the end of this SE). The NRC staff performed independent confirmatory dose evaluations and audited the licensee's analyses to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident meet the applicable accident dose acceptance criteria, as listed in Table 1, and are, therefore, acceptable.

3.1.4 Main Steam Line Break

MSLB Pre-Accident Iodine Spike Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.0019	25	0.0009	25	0.0012	5
Thyroid	1.2	300	0.61	300	1.0	50
Beta					0.035	50

MSLB Concurrent Iodine Spike Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.0038	2.5	0.0044	2.5	0.0014	5
Thyroid	1.4	30	2.5	30	2.5	50
Beta					0.036	50

The licensee evaluated the radiological consequences of an MSLB accident as a part of the implementation of RG 1.195 and the replacement of CPSES Unit 1 SGs. The MSLB accident considered is the complete severance of the largest main steam line outside containment. The radiological consequences of an MSLB break outside containment will bound the radiological consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences.

The licensee's evaluation indicates that no fuel damage is predicted as a result of an MSLB accident. Therefore, consistent with the current licensing analysis basis and RG 1.195, the licensee performed the MSLB accident analyses assuming that the accident occurs at full power. The current CPSES TSs limit the RCS iodine activity to 0.45 µCi/gm DE I-131. The analyses presented in this LAR support returning the TS limit to 1.0 µCi/gm DE I-131. The SG secondary coolant activity is based on 0.1 µCi/gm DE I-131. As in the SGTR accident, the licensee's MSLB evaluation includes the effects of primary system iodine spiking for both the pre-accident iodine spike case and the concurrent iodine spike case. The spiking cases are as described for the SGTR with the following exception. For the MSLB accident, the concurrent iodine spike is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. In effect, it is assumed that the MSLB accident results in a more severe concurrent iodine spike than the SGTR accident.

The MSLB accident begins with a break in one of the main steam lines leading from an SG to the turbine. The SG that experiences a secondary side depressurization as a result of an

MSLB is referred to as being in a faulted condition. The licensee uses the term “affected SG” to describe the faulted SG. In order to maximize the CR dose, the licensee assumed that the steam line break occurs in the turbine building. The affected SG is assumed to release steam for 20 hours, which is the time required for the RCS to be cooled down to 212 °F. The 20-hour steaming period is based on the time necessary to cooldown to 212 °F, crediting safety grade equipment only.

The licensee evaluated the accident assuming a concurrent LOOP. Due to the assumption of a LOOP, the condenser is unavailable and cooldown of the primary system is accomplished through the release of steam from the intact SG ARVs.

For the affected SG, TXU assumed the release passes directly into the turbine building with no credit taken for holdup, partitioning, or scrubbing by the SG liquid. The licensee did not take credit for any holdup or dilution in the turbine building. The TXU analysis assumes the release into the turbine building is exhausted to the environment and subsequently transported from the environment into the CR assuming conservative atmospheric dispersion factors.

The near instantaneous release of the secondary coolant from the affected SG represents a significant contribution to the total dose from an MSLB, since the inventory is evaluated at 0.1 μCi/gm DE I-131. The licensee conservatively assumed that during the first 20 hours, primary coolant leaks into the affected SG at the rate of 500 gpd (0.35 gpm) directly releasing all of the coolant activity to the environment. This release is assumed to continue for 20 hours, until the RCS has cooled to below 212 °F, at which time the release from this pathway terminates.

The licensee assigned 0.6528 gpm primary-to-secondary side leakage to the three intact SGs. The licensee assumed that this leakage continues for 8 hours.

An assumed iodine partition factor in the unaffected SGs results in 1 percent of the iodines in the SG bulk liquid being released to the environment at the steaming rate. Radionuclides initially in the steam space do not provide any significant dose contribution. The transport to the environment of noble gases from the primary coolant is assumed to occur without any mitigation or holdup.

3.1.4.1 Control Room and Atmospheric Dispersion Modeling

The licensee’s modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.4.2 MSLB Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and GDC 19, and the accident-specific dose acceptance criteria in RG 1.195. The NRC staff’s review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in RG 1.195. The assumptions found acceptable to the NRC staff are presented in Tables 8 and 1 (both tables are at the end of this SE). The NRC staff performed independent confirmatory dose evaluations and audited the licensee's analyses to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the MSLB meet the applicable accident dose acceptance criteria as listed in Table 1 and are, therefore, acceptable.

3.1.5 Primary Coolant Pump Locked Rotor Accident

LRA Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.12	2.5	0.037	2.5	0.14	5
Thyroid	1.5	30	2.5	30	1.6	50
Beta					1.8	50

The accident considered begins with the instantaneous seizure of a reactor coolant pump rotor, which causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow while the reactor is at power causes a degradation of core heat transfer, resulting in localized temperature and pressure changes in the core. As a result, the licensee assumes that fuel cladding damage occurs due to a departure from nucleate boiling. Activity from the fuel damage is transported to the secondary side due to primary-to-secondary side leakage evaluated at the TS limit. It is assumed that the LRA does not cause an increase in the magnitude of the pre-existing primary-to-secondary leakage. The licensee analyzed the LRA accident in this LAR because the dose analysis assumptions were not clearly bounded by the CREA.

As a result of the LRA, the licensee has determined that 10 percent of the core fuel inventory gap activity would be released to the RCS. The radioactivity in the RCS is assumed to be transported to the secondary side with primary-to-secondary leakage of 1 gpm total. The licensee assumed that the release continues for 8 hours after which RHR is placed in service

The licensee used the RADTRAD 3.03 computer code to model the time dependent transport of radionuclides, from the primary-tosecondary side and consequently to the environment via the ARVs. The licensee's analysis conforms with Appendix G of RG 1.195, which identifies acceptable radiological analysis assumptions for an LRA.

3.1.5.1 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.5.2 LRA Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and GDC 19 and the accident-specific dose acceptance criteria in RG 1.195. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in RG 1.195. The assumptions found acceptable to the NRC staff are presented in Tables 9 and 1 (both tables are at the end of this SE). The NRC staff performed independent confirmatory dose evaluations and audited the licensee's analyses to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the LRA were found to meet the applicable accident dose acceptance criteria as listed in Table 1 and are, therefore, acceptable.

3.1.6 Control Rod Ejection Accident

Rod Ejection: Containment Release Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.059	6.3	0.031	6.3	0.02	5
Thyroid	14	75	20	75	16	50
Beta					0.34	50

Rod Ejection: Secondary Release Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.29	6.3	0.091	6.3	0.34	5
Thyroid	2.3	75	3.8	75	2.4	50
Beta					4.3	50

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a rod control cluster assembly and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion, together with an adverse core power distribution. For this accident, localized damage to fuel cladding and a limited amount of fuel melt are projected. The licensee assumed that as a result of localized fuel cladding damage, 10 percent of the gap activity is released to the primary coolant. In addition, the licensee assumes that 0.25 percent of the fuel inventory is also released to the primary coolant as a result of limited fuel melting. The mechanical failure breaches the reactor pressure vessel head resulting in a release of primary coolant to the containment atmosphere. Releases to the environment are assumed to occur through two separate pathways:

- Release of containment atmosphere (using design leakage assumptions), and

- Release of RCS inventory via primary-to-secondary leakage through SGs.

To evaluate the release to containment atmosphere, the licensee employed the guidance from Appendix H of RG 1.195. TXU assumed that 10 percent of the fuel rods in the core fail cladding, releasing the fission product inventory in the fuel rod gap. The licensee assumed that 10 percent of the core inventory of iodines and noble gases is in the fuel rod gap. Therefore, for the fuel clad failure, the fraction of core activity released is 0.01 for both iodines and noble gases. In addition, the licensee assumed that localized heating causes 0.25 percent of the fuel to melt, releasing 25 percent of the iodines and 100 percent of the noble gases contained in the melted fuel. As a result of the fuel melt portion of the fuel damage, the fraction of the core halogen activity released is 0.000625 (0.0025 x 0.25) and the fraction of noble gas activity released is 0.0025. The total activity released as a result of the fuel damage from the CREA is the sum of the clad failure fraction and the fuel melt fraction.

For the first release case through containment leakage, the radioactivity release from the fuel as described above was assumed to be released instantaneously into the containment. TXU has determined that containment sprays will not initiate due to a CREA and, as a result, the licensee did not evaluate dose contributions from ECCS leakage and RWST back leakage as in the LOCA analysis. The containment is assumed to leak at the TS leak rate of $L_{a.}$ The licensee assumed that the containment leak rate is reduced by 50 percent at 24 hours for both the offsite and the CR analyses.

The second release path evaluated by the licensee is via the secondary system. The licensee based the evaluation of the activity in the secondary system release on the guidance in Appendix H of RG 1.195. The core release fractions for iodines and noble gases are based on the assumed consequences of 10-percent failed fuel and 0.25-percent melted fuel, as in the containment release case. To evaluate the fuel clad failure portion of the fuel damage, the fraction of core activity released is 0.01 for both iodines and noble gases as in the containment release case. For the secondary release pathway, the licensee assumed that 50 percent of the iodines and 100 percent of the noble gases contained in the melted fuel are released to the RCS. Therefore, as a result of the fuel melt portion of the fuel damage the fraction of the core halogen activity released to the RCS is 0.00125 (0.0025 x 0.5) and the fraction of noble gas activity released is 0.0025.

For the secondary system release case, the licensee assumed that fission products released from the fuel are instantaneously and homogeneously mixed in the RCS and transported to the secondary side of the SGs via primary-to-secondary leakage at the TS value of 1 gpm for 8 hours. The licensee has determined that, for this event, an 8-hour time period is required for the primary system pressure to fall below the secondary side system pressure. A LOOP is conservatively assumed to occur at T = 0, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment via the relief valves. During the first 8 hours of the accident the only steam release is assumed to be via the secondary safety valve. When the primary system pressure drops below the secondary side pressure, the relief valve closes. The licensee assumed the chemical form of the iodines released from the SGs to be 97-percent elemental and 3-percent organic as is consistent with the RG 1.195. As in the evaluation of the MSLB accident, the licensee assumed an iodine partition factor of 100 in the SGs and assumed that the noble gas activity released to the secondary system is released to the environment without reduction or mitigation.

3.1.6.1 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.6.2 CREA Conclusion

The licensee evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 100.11 and GDC 19 and the accident-specific dose acceptance criteria in RG 1.195. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in RG 1.195. The assumptions found acceptable to the NRC staff are presented in Tables 9 and 1 (both tables are at the end of this SE). The NRC staff performed independent confirmatory dose evaluations and audited the licensee's analyses to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the CREA were found to meet the applicable accident dose acceptance criteria as listed in Table 1 and are, therefore, acceptable.

3.1.7 <u>CVCS Letdown Line Break</u>

CVCS Letdown Line Break Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.039	2.5	0.0058	2.5	0.021	5
Thyroid	5.7	30	0.85	30	0.64	50
Beta					0.54	50

This event assumes a complete severance of the 3-inch CVCS letdown line just outside containment, between the outboard letdown isolation valve and letdown heat exchanger, at full-rated power condition. The severance of the letdown line results in a loss of reactor coolant at the rate of about 190 gpm, which is in the makeup capacity of any two of the three charging pumps, as found acceptable in the CPSES current licensing basis. The licensee assumes that 20.1 percent of the leaking coolant flashes to steam and that all of the iodine in this steam is assumed to become airborne and is available for release to the atmosphere. Also, all noble gases contained in the leaking primary coolant are available for release to the atmosphere. The time required for the operator to identify the accident and initiate the closure of the letdown isolation valve is expected to be within 30 minutes after accident initiation including 10 seconds for the letdown isolation valve closure time.

This event is not listed as an accident in RG 1.195, but is discussed in SRP 15.6.2 "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," where the appropriate accident specific dose acceptance criteria are given. These offsite dose criteria are a small fraction (i.e. 10 percent) of the 10 CFR 100.11 limits. In its reanalysis the licensee used the assumptions contained in the current licensing basis as discussed in CPSES FSAR. The licensee performed this analysis using RADTRAD 3.03 to ensure a consistent licensing basis for all accidents. The NRC staff performed a detailed audit of the TXU calculations. These calculations were not submitted with the LAR, but were made available for review by the NRC staff.

3.1.7.1 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.7.2 CVCS Letdown Line Break Conclusion

The licensee evaluated the radiological consequences resulting from the postulated CVCS letdown line break and concluded that the radiological consequences at the CR, EAB, and LPZ are within the dose guidelines provided in GDC 19 and a small fraction of 10 CFR 100.11. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in SRP 15.6.2. The NRC staff performed an independent audit to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the CVCS letdown line break were found to meet the applicable accident dose acceptance criteria as listed in Table 1 and are, therefore, acceptable.

3.1.8 <u>Gas Decay Tank Rupture</u>

Gas Decay Tank Rupture Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.19	0.50	0.028	0.50	0.27	5
Thyroid	N/A	N/A	N/A	N/A	N/A	50
Beta					7.7	50

The gas decay tank rupture is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste decay tank as a consequence of a failure of a single gas decay tank or associated piping.

This event is not listed as an accident in RG 1.195, but is discussed in Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," which is contained in SRP 11.3, "Gaseous Waste Management Systems," where the appropriate accident-specific dose acceptance criteria are given. The offsite dose criterion is that the calculated whole body dose is substantially below the 10 CFR 100.11 limits (i.e., 0.5 rem). In its reanalysis the licensee used the assumptions contained in the current licensing basis as discussed in CPSES FSAR. The licensee performed this analysis using RADTRAD 3.03 to ensure a consistent licensing basis for all accidents. The NRC staff performed a detailed audit of the TXU calculations. These calculations were not submitted with the LAR, but were made available for review by the NRC staff.

3.1.8.1 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively.

3.1.8.2 Gas Decay Tank Rupture Conclusion

The licensee evaluated the radiological consequences resulting from the postulated gas decay tank rupture and concluded that the radiological consequences at the CR and EAB are within the dose guidelines provided in GDC 19 and meet the dose criterion offsite, respectively. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in SRP 11.3. The NRC staff performed an independent audit to ensure a thorough understanding of the licensee's methods. The EAB and CR doses estimated by the licensee for the gas decay tank rupture were found to meet the applicable accident dose acceptance criteria as listed in Table 1 and are, therefore, acceptable.

3.1.9 Liquid Waste Tank Rupture

Liquid Waste Tank Rupture Dose Results						
Dose (rem)	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
WB	0.0038	0.50	0.00057	0.50	0.0036	5
Thyroid	2.3	6.0	0.34	6.0	41	50
Beta					0.023	50

The liquid waste tank rupture is defined as the uncontrolled atmospheric release from the 30,000 gallon floor drain tank due to the postulated rupture of the tank. This tank is assumed to be 80-percent full of reactor coolant. The entire content of the tank is assumed to be released to the building. The activity released from the tank is assumed to be released to the atmosphere over a 2-hour period at ground level.

This event is not listed as an accident in RG 1.195, but is discussed in SRP 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures." In its reanalysis, the licensee used the assumptions contained in the current licensing basis as discussed in CPSES FSAR. The offsite dose criterion remain unchanged from the current licensing basis. The licensee performed this analysis using RADTRAD 3.03 to ensure a consistent licensing basis for all accidents. The NRC staff performed a detailed audit of the TXU calculations. These calculations were not submitted with the LAR, but were made available for review by the NRC staff.

3.1.9.1 Control Room and Atmospheric Dispersion Modeling

The licensee's modeling of the CR and the atmospheric dispersion modeling are discussed below in Sections 3.2 and 3.3 of this SE, respectively. The CR does not isolate during this accident, and the licensee does not credit manual operator action for isolation.

3.1.9.2 Liquid Waste Tank Rupture Conclusion

The licensee evaluated the radiological consequences resulting from the postulated liquid waste tank rupture and concluded that the radiological consequences at the CR, EAB, and LPZ are within the dose guidelines provided in GDC 19 and meet the dose criterion offsite, respectively. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance in SRP 15.7.3 and the CPSES current licensing basis. The NRC staff performed an independent audit to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the liquid waste tank rupture were found to meet the applicable accident dose acceptance criteria as listed in Table 1 and are, therefore, acceptable.

3.2 Control Room Habitability

At the start of events 1 through 9, the CR air-conditioning system is assumed to be in normal mode. An SI actuation signal or a high-radiation signal from the CR air intake monitors will isolate the CR and initiate the CR emergency filtration system (CREFS) in the recirculation mode of operation. Filtered inflow, filtered recirculation flow, damper leakage, and unfiltered inleakage into the CR are considered.

TXU used current CPSES FSAR values for CREFS operating mode sequence and flow rates with one exception. The only licensing basis change to the CR assumptions for this LAR is the licensee's unfiltered inleakage value. It is conservatively increased from 12 cubic feet per minute (cfm) to 27 cfm. The current value is based upon 10 cfm for CR ingress/egress and 2 cfm for leakage from ductwork passing through the CR boundary. The additional 15 cfm, in the revised value, is leakage from other sources. The licensee determined through testing in accordance with NRC Generic Letter 2003-01, "Control Room Habitability," that the actual CR inleakage value is 0 cfm.

This CR modeling was used in the analysis of each of the nine events discussed above.

3.3 Atmospheric Dispersion Factors

The licensee used previously generated atmospheric dispersion factor (χ/Q) values for the CR, EAB, and LPZ. Since these values are already part of the facilities licensing basis, and since there are no release locations or receptors different from those in the current licensing basis, the NRC staff found them acceptable for use in the DBA dose assessments performed in support of this LAR.

3.4 Conclusions of Technical Evaluation

This licensing action is considered an implementation of RG 1.195 and will supersede corresponding radiological analysis assumptions provided in other regulatory guides when used in conjunction with guidance that is in RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors." TXU submitted LAR 06-011 on October 23, 2006, through which the implementation of RG 1.196 is to occur. The NRC staff finds the implementation of RG 1.195 to be acceptable contingent upon the acceptability of CPSES's implementation of RG 1.196. The licensee has committed in LAR 06-011, dated October 23, 2006, to implement TSTF-448 to complete the CR habitability program requirements in RG 1.196. The NRC staff's current perspective is that TSTF-448 implementation contained in LAR 06-011 will likely be approved in accordance with the Consolidated Line Item Improvement Process (CLIIP).

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with implementation of RG 1.195 and Unit 1 SG replacement at CPSES. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff further finds reasonable assurance that the licensing basis DBA analyses for CPSES, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression, and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to 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 published November 8, 2005 (70 FR 67754). 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.

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Table 1

CPSES Licensee Calculated Radiological Consequences

Accident and Dose Type	Dose (rem)					
	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
LOCA						
WB	0.79	25	0.29	25	2.3	5
Thyroid	63.0	300	47.0	300	43	50
Beta					16.0	50
FHA						
WB	0.087	6.3	0.013	6.3	0.13	5
Thyroid	22	75	3.3	75	3.6	50
Beta					4.1	50
SGTR Pre-Accident Iodine Spike						
WB	0.08	25	0.013	25	0.07	5
Thyroid	22.0	300	3.4	300	18	50
Beta					1.9	50
SGTR Concurrent Iodine Spike						
WB	0.077	2.5	0.012	2.5	0.069	5
Thyroid	8.8	30	1.4	30	2.0	50
Beta					1.9	50
MSLB Pre-Accident Iodine Spike						
WB	0.0019	25	0.0009	25	0.0012	5
Thyroid	1.2	300	0.61	300	1.0	50
Beta					0.035	50

Table 1 (Cont.)

CPSES Licensee Calculated Radiological Consequences

Accident and Dose Type	Dose (rem)					
	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
MSLB Concurrent Iodine Spike						
WB	0.0038	2.5	0.0044	2.5	0.0014	5
Thyroid	1.4	30	2.5	30	2.5	50
Beta					0.036	50
Locked Rotor						
WB	0.12	2.5	0.037	2.5	0.14	5
Thyroid	1.5	30	2.5	30	1.6	50
Beta					1.8	50
Rod Ejection: Containment Release						
WB	0.059	6.3	0.031	6.3	0.02	5
Thyroid	14	75	20	75	16	50
Beta					0.34	50
Rod Ejection: Secondary Release						
WB	0.29	6.3	0.091	6.3	0.34	5
Thyroid	2.3	75	3.8	75	2.4	50
Beta					4.3	50
CVCS Break						
WB	0.039	2.5	0.0058	2.5	0.021	5
Thyroid	5.7	30	0.85	30	0.64	50
Beta					0.54	50

Table 1 (Cont.)

CPSES Licensee Calculated Radiological Consequences

Accident and Dose Type	Dose (rem)					
	2-hr EAB CPSES	2-hr EAB Criterion	30-day LPZ CPSES	30-day LPZ Criterion	MCR CPSES	MCR Criterion
Gas Decay Tank Rupture						
WB	0.19	0.50	0.028	0.50	0.27	5
Thyroid	N/A	N/A	N/A	N/A	N/A	50
Beta					7.7	50
Liquid Waste Tank Rupture						
WB	0.0038	0.50	0.00057	0.50	0.0036	5
Thyroid	2.3	6.0	0.34	6.0	41	50
Beta					0.023	50

Table 2

CPSES Atmospheric Dispersion Factors

Receptor / Duration	χ/Q (sec/m^3)
Control Room, from Containment Leakage	
0 - 8 hours	3.04E-03
8 - 24 hours	1.82E-03
24 - 96 hours	6.08E-04
96 - 720 hours	1.4E-04
Control Room, from ESF Equipment Leakage Outside Containment/Atmospheric Relief Valves	
0 - 8 hours	2.96E-03
8 - 24 hours	1.86E-03
24 - 96 hours	6.51E-04
96 - 720 hours	1.78E-04
EAB	
0 - 2 hours	1.6E-04
LPZ	
0 - 8 hours	2.4E-05
8 - 24 hours	1.6E-05
24 - 96 hours	6.2E-06
96 - 720 hours	1.7E-06

Table 3

CPSES Control Room Data and Assumptions

CR effective volume		423,032 ft^3
Normal CR intake flow rate prior to isolation		3000 cfm
Unfiltered inleakage		27 cfm
Air intake flow rate (filtered) (emergency recirculation mode)		800 cfm
Air intake flow rate (emergency ventilation mode)		3800 cfm
Emergency Ventilation System Recirculation flow rate (emergency recirculation mode)		7200 cfm
Emergency Ventilation System Recirculation flow rate (emergency ventilation mode)		4200 cfm
Response time for CR to Isolate upon receipt of Control Room Ventilation Radiation Monitor Alarm Signal		30 seconds
Filter Efficiencies for the Emergency Ventilation System		99% elemental 99% aerosol 99% organic
Control building wall thickness		2 ft concrete
Control room ceiling/roof thickness		3 ft concrete
CR occupancy factors	0 - 24 hours	1.0
	24 - 96 hours	0.6
	96 - 720 hours	0.4
Breathing rate for CR dose analyses		3.5E-04 m^3/sec

Table 4

CPSES Data and Assumptions for the LOCA

Containment free air volume	3.031E+06 ft^3
Containment leak rate	0.1 weight percent per day
Containment leak rate reduction	50% after 24 hours
Iodine chemical form in containment atmosphere	91% elemental iodine 4% organic iodine 5% particulate iodine
Iodine chemical form in the sump	97% elemental 3% organic
Containment sump pH	≥ 7
Containment Spray System (CSS) effective operation period	74.3 sec for full flow
Elemental iodine removal coefficient	10 hr^{-1} until a DF of 100 then 0
Particulate iodine removal coefficient	11.4 hr^{-1} until a DF of 50 1.14 hr^{-1} at a DF of > 50
Duration of elemental iodine removal effectiveness	2.518 hr (DF of 100 attained)
Duration of particulate iodine removal effectiveness	2.03 hr (DF of 50 attained) 4.0 hr (sprays assumed to be terminated)
CSS containment coverage volume	1,705,945 ft^3
ECCS leakage outside containment	2 gpm
Minimum available RWST volume	4.053E+05 gallons (1.543E+09 cc)

Table 5

CPSES Data and Assumptions for the FHA

Fuel clad damage	All rods in 1 assembly
Gap fractions	
I-131	8%
Kr-85	10%
Other noble gases	5%
Other iodines	5%
Pool DF	
Noble gases	1
Iodines	160 (effective DF)
Release point	Ground level
Decay time	100 hours
Radial peaking factor	1.65
Duration of release	2 hours

Table 6

CPSES Data and Assumptions for the SGTR Accident

Primary-to-secondary leak rate TS limit	150 gpd (to intact SGs)
Secondary iodine TS limit	0.1 µCi/gm DE I-131
RCS limit for normal operation	
Gross gamma	100/ E_{bar}
Iodine	1.0 µCi/gm DE I-131
RCS limit for pre-accident iodine spike	60 µCi/gm DE I-131
Coincident spike appearance rate multiplier	335
Iodine spike duration	5.75 hours
LOOP	Coincident with release
Release points	ARV
Secure release from affected SG	13 minutes 30 minutes (shut failed open ARV)
Secure release from intact SGs	8 hours (RHR intiated)
Chemical form of iodine released from SGs	Elemental 97% Organic 3%
Iodine partition factor	0.01
Condenser partition factor	0.15
Duration of release to environment	
Intact SGs	0 - 8 hours
Affected SG	0 - 30 minutes
Initial Affected SG Mass	4.18E+07 grams
Initial Intact SG Masses	1.25E+08 grams

Table 7

CPSES Data and Assumptions for the MSLB Accident

Primary-to-secondary leak rate TS limit	150 gpd (to intact SGs)
Secondary iodine TS limit	0.1 μCi/gm DE I-131
RCS limit for normal operation	
Gross gamma	100/ E_{bar}
Iodine	1.0 μCi/gm DE I-131
RCS limit for pre-accident iodine spike	60 μCi/gm DE I-131
Coincident spike appearance rate multiplier	500
LOOP	Assumed to occur at accident initiation
Release points:	
Affected SG	Turbine building
Intact SG	SG ARVs
Iodine partition factor for Affected SG	1.0
Iodine partition factor for intact SGs	0.01
Primary-to-secondary leakage	
Affected SG	500 gpd
Intact SG	0.6528 gpm
Total	1 gpm
SG liquid mass	164,200 lbm
Duration of SG release:	
Affected SG	20 hours
Intact SG	8 hours
Steam release from affected SG	
Initial inventory	35,200 lbm/min (0 - 5 min)
Primary-to-secondary leak	2.9 lbm/min (0 - 20 hrs)
Steam Release from intact SGs	
0 - 2 hours	4.38E+05 lbm/min
2 - 8 hours	9.60E+05 lbm/min

Table 8

CPSES Data and Assumptions for the LRA

Fuel clad failure	10%
RCS iodine limit for normal operation	1.0 μCi/gm DE I-131
Secondary iodine TS limit	0.1 μCi/gm DE I-131
Radial peaking factor	1.65
Primary-to-secondary leak rate	1 gpm
Release points	SG ARV
Chemical form of iodine released from the SGs to the environment	3% organic iodide 97% elemental iodine
Fraction of fission product inventory in gap	
I-131	0.08
iodines	0.05
Noble gases	0.05
Iodine partition fraction	0.01
Release Duration	8 hours (RHR placed in service)
Total mass of steam to atmosphere from intact SGs	
0 - 2 hours	450,000 lbm
2 - 8 hours	1,002,000 lbm
Initial SG liquid mass	1.67E+08 grams

Table 9

Data and Assumptions for the CREA

Containment free air volume	3.031E+06 ft^3
Fuel clad failure	10%
Fraction of core inventory in gap	
Noble gasses	10%
Iodine	10%
Core fuel melt	0.25%
Release fractions for melted fuel	
Containment release	
Noble gasses	100%
Iodines	25%
Reactor coolant release	
Noble gasses	100%
Iodines	50%
Chemical form of iodine released from the SG to the environment	3% organic iodide 97% elemental iodine
Total primary-to-secondary leakage through all SGs	1 gpm
Time for primary system pressure to fall below secondary system pressure	8 hours
Duration of steam releases	8 hours
Steam released from 0 to 8 hours	1,452,000 lbm
Release point	SG ARV

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