Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498 and STN 50-499
Proposed License Amendment Concerning Radiological Aspects of Operation at Reduced Feedwater Temperature and of Operation with Replacement Steam Generators

Reference: 1) Letter from L. E. Martin to U.S. Nuclear Regulatory Commission dated May 7, 1998, (ST-NOC-AE-00159)

The South Texas Project Nuclear Operating Company (STPNOC) proposes to amend the South Texas Project Operating Licenses NPF-76 and NPF-80 by incorporating the attached changes into the Updated Final Safety Analysis Report (UFSAR) for South Texas Project Units 1 and 2 The purpose of this license change is to revise the offsite dose licensing basis to account for operation of the existing steam generators at reduced feedwater inlet temperatures, and to account for operation of the new replacement steam generators (The existing Unit 1 Westinghouse Model E steam generators are currently planned to be replaced with Westinghouse Model Δ94 steam generators in May, 2000). The proposed changes in this submittal include revised calculated offsite dose rates for three existing UFSAR Chapter 15 analyzed accidents and inclusion of a discussion in Chapter 15 of the radiological analysis for the voltage-based repair criteria for steam generator tubes.

Current South Texas Project licensing basis calculations for offsite dose consequences from UFSAR Chapter 15 analyzed accidents are based on the assumption of operation with existing Model E steam generators at a nominal feedwater inlet temperature of 440°F. However, South Texas Project proposes to operate with existing Model E steam generator feedwater inlet temperatures as low as 420°F to achieve 100% reactor power during degraded steam generator conditions, and to replace the Model E steam generators with Model Δ94 steam generators (which will operate with feedwater inlet temperatures as low as 390°F). As a result of these alternate modes of steam generator operation, the radiological analyses presented in Chapters 11 and 15 were evaluated and revised, as necessary. This opportunity was also taken up update

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methodologies and improve analytical assumptions in the revised analyses. Primarily as a result of the changes in methodologies and assumptions, the offsite dose consequences of three UFSAR analyzed accidents increased and therefore require review as unreviewed safety questions. The enclosed safety evaluation shows that the increase does not constitute a significant hazard.

Please note that other proposed changes to the operating license, pertaining to the Δ94 steam generators, have been previously submitted for Nuclear Regulatory Commission approval. A general description of these additional proposed changes, as well as other submittals to support licensing for the Δ94 steam generators, is contained in Reference 1.

The South Texas Project has reviewed the attached proposed amendment in accordance with 10CFR50.92 and has determined that the amendment does not involve a significant hazards consideration. Additionally, it has been determined that the proposed amendment satisfies the criteria of 10CFR51.22(c)(9) for categorical exclusion from the requirement for an environmental assessment. The South Texas Project Plant Operations Review Committee has reviewed the proposed amendment and recommended its approval. Also, the South Texas Project Nuclear Safety Review Board has reviewed and approved the proposed amendment.

The required affidavit, along with a Safety Evaluation and No Significant Hazards Consideration Determination associated with the proposed amendment, and the marked-up Updated Final Safety Analysis Report pages are included as attachments to this letter.

The South Texas Project requests that the effective date of this amendment be 30 days after the date of Nuclear Regulatory Commission approval. Although this request is neither exigent nor an emergency, issuance of this amendment by the Nuclear Regulatory Commission by June 1, 1999, is requested.

The South Texas Project is providing the State of Texas with a copy of this proposal in accordance with 10CFR50.91(b). Also, it has been determined that there are no new licensing commitments contained in this document.

If there are any questions regarding this proposed amendment, please contact either Mr. M. A. McBurnett at (512) 972-7206 or me at (512) 972-8787.

T. H. Cloninger
Vice President,
Engineering and Technical Services

BJS/

Attachments:

1. Affidavit
2. Summary and Description of the Proposed Changes
3. Determination of No Significant Hazards Consideration
4. Updated Final Safety Analysis Report Marked-Up Pages

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ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter	)	
	)	
STP Nuclear Operating Company	)	Docket Nos. 50-498
	)	50-499
South Texas Project Units 1 & 2	)	

AFFIDAVIT

I, T. H. Cloninger, being duly sworn, hereby depose and say that I am the Vice President, Engineering and Technical Services of the South Texas Project; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached proposed amendment to South Texas Project Operating Licenses NPF-76 and NPF-80; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

T. H. Cloninger
Vice President,
Engineering and Technical Services

STATE OF TEXAS	)
	)
COUNTY OF MATAGORDA	)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 29th day of September, 1998.

NOTARY PUBLIC STATE OF TEXAS
LINDA RITTENBERRY
Notary Public, State of Texas
My Commission Expires
OCT. 9, 2001

Linda Rittenberry
Notary Public in and for the
State of Texas

ATTACHMENT 2

SUMMARY AND DESCRIPTION OF THE PROPOSED CHANGES

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1. SUMMARY

The South Texas Project (STP) requests approval to increase the calculated offsite dose consequences of the following accidents described in (or added to) Chapter 15 of the Updated Final Safety Analysis Report (UFSAR):

- Main Steamline Break (UFSAR Section 15.1.5)
- Voltage-Based Repair Criteria Dose Analysis (new UFSAR Section 15.1.6)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR Section 15.3.3)
- Rod Cluster Control Assembly (RCCA) Ejection Accident (UFSAR Section 15.4.8)

The increases in offsite dose are minimal and all doses remain below the dose limits for the respective accidents, as specified by Title 10 Code of Federal Regulations Section 100 (10 CFR 100) and the Standard Review Plan (NUREG-0800).

The calculated increase results primarily from assumption and modeling changes in the calculation methods. The changes to the calculation also include the affects of the below two items. Although not specifically quantified, their effects would not be expected to be significant.

1. Reduction of the inlet feedwater temperature for the existing Model E steam generators from 440°F to 420°F, and
2. Replacement of the current Westinghouse Model E steam generators with Westinghouse Model Delta 94 (referred to as 'Δ94') steam generators.

Information concerning postulated accidents which remain bounding by the current UFSAR analyses is presented in this submittal.

This submittal only addresses the radiological aspects of these proposed changes. Other aspects of the proposed reduction in feedwater temperature are being addressed by the 10 CFR 50.59 review process.

The radiological impacts of these proposed changes to the facility, based on the results presented in the UFSAR, are minimal. Consequently, the South Texas Project requests that the Nuclear Regulatory Commission (NRC) approve the proposed changes.

2. BACKGROUND

As steam generators wear through use, steam generator tubes have a greater likelihood to crack and leak. When steam generator tube inspections detect a cracked tube, the tube is generally plugged to prevent reactor coolant leakage. However, plugging a steam generator tube also prevents that tube from transferring reactor coolant heat to the secondary system. As more and more steam generator tubes are plugged, the steam generator loses its ability to transfer design reactor coolant heat to the secondary system, resulting in reduced secondary system steam pressure. This condition is referred to as 'steam generator degradation'.

The South Texas Project is pursuing the following two proposed modifications to the facility:

1. Reduction of inlet feedwater temperature for the existing Model E steam generators from 440°F to 420°F.

   The purpose of this change is to allow feedwater inlet temperature to the existing steam generators (SGs) to be operated in a range between 440°F and 420°F. The feedwater temperature reduction will allow 100% reactor power to be achieved with degraded steam generators.

   Due to steam generator degradation, the effluent steam pressure has decreased following every refueling outage. After 1RE06 (ending June, 1996), the turbine governor valves were wide open prior to reaching 100% reactor power (i.e., due to reduced secondary steam pressure, the turbine generator could only carry an electrical load equivalent to less than 100% reactor power). However, the reduced reactor power condition was minimal and was for a limited duration. Based on these declining pressure trends, subsequent startups following refueling outages, until steam generator replacement, will likely result in greater reductions of reactor power. This is assuming there is no adjustment of Tavg to increase steam pressure or any other abatement measures.

   Reducing the steam generator feedwater inlet temperature will increase reactor power by reducing reactor coolant system (RCS) T-cold. On the secondary side, as the feedwater heaters are partially bypassed, there will be less water available in the feedwater heater to condense High Pressure (HP) turbine extraction steam. This excess steam will then be forced through the remainder of the HP turbine, Moisture Separator Reheater (MSR), and Low Pressure (LP) turbines thereby increasing power output. The overall effect is to produce slightly more electrical power at a reduced efficiency.

2. Replacement of the current Model E steam generators with Δ94 steam generators.

   This modification is necessary due to the degraded condition of the current Model E steam generators in Unit 1. The radiological analyses performed assume the inlet feedwater temperature for the Δ94 steam generators may be as low as 390°F.

The radiological impacts of these proposed modifications on the facility were determined to be minimal. Also, due to timing considerations of the current replacement steam generator project in Unit 1, and the desire to have the ability to reduce the feedwater temperature in either unit, bounding accident analyses were performed for offsite dose consequences. Table 2.0-1 provides a summary of the results. In addition, the analysis methodologies were revised for several accidents. These methodology improvements are responsible for the majority of the dose increases presented.

Table 2.0-1
Summary of Radiological Impacts

UFSAR Section	Topic	Impact	Comments
11.1	Primary Systems and Secondary Systems Isotopic Content	Negligible	
11.2	Liquid Radwaste System Isotopic Content	Negligible	
11.3	Gaseous Radwaste System Isotopic Content	Negligible	
15.1.5	Main Steamline Break	Offsite dose increase	Doses increase as a result of methodology change.
15.1.6 (new section)	Voltage-Based Repair Criteria Dose Analysis	Increase in the allowed post-MSLB accident primary-to-secondary leakage.	Doses maintained at 90% of acceptance criteria.
15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Current UFSAR analysis remains bounding for thyroid doses. Proposed changes result in small increase in offsite whole body and skin doses.	Existing errors for the Low Population Zone (LPZ) thyroid doses in Table 15.3-4 are corrected.
15.4.8	RCCA Ejection Accident	Minimal increase in offsite whole body and skin doses.	
15.6.3	Steam Generator Tube Rupture	Current UFSAR analysis remains bounding.	
15.6.5	Loss of Coolant Accident	Current UFSAR analysis remains bounding.	
15.7.1	Gaseous Radwaste Processing System Tank Failure	Negligible	
15.7.2 15.7.3	Liquid Radwaste Processing System Tank Failure	Negligible	

3. DESCRIPTION OF CHANGES

The following sections of the UFSAR have been updated (or newly added) to reflect the revised analyses:

3.1 Chapter 11 Table of Contents

The Table of Contents is updated to include the new Sections 11.1.7 and 11.1.8.

3.2 Primary Systems and Secondary Systems Isotopic Analysis

New Section 11.1.7 is added to state that a reduction in feedwater temperature to 420°F in Model E steam generators will cause a negligible change in the primary and secondary system isotopics. New Section 11.1.8 is added to state that replacement of the Model E steam generators by Δ94 steam generators, operating at a feedwater temperature as low as 390°F, will cause a negligible change in the primary system isotopics.

3.3 Liquid Radwaste Processing System Isotopic Content

A statement is added to Section 11.2.1 to state that a reduction in feedwater temperature to 420°F in Model E steam generators or replacement of the Model E steam generators by Δ94 steam generators, operating at a feedwater temperature as low as 390°F, will have a negligible impact on the isotopic inventory of the liquid waste processing system and the radiological consequences of a Liquid Waste Processing System (LWPS) failure, as described in Chapter 15.7.

3.4 Gaseous Radwaste Processing System Isotopic Content

A statement is added to Section 11.3.1 to state that a reduction in feedwater temperature to 420°F in Model E steam generators or replacement of the Model E steam generators by Δ94 steam generators, operating at a feedwater temperature as low as 390°F, will have a negligible impact on the isotopic inventory of the gaseous waste processing system and the radiological consequences of a Gaseous Waste Processing System (GWPS) failure, as described in Chapter 15.7.1.

3.5 Chapter 15 Table of Contents

The Table of Contents is updated to include the new Sections 15.1.6, 15.A.6, 15.A.7, and 15.B.4.

3.6 Chapter 15 List of Tables

The List of Tables is updated to include the new Tables 15.1-4 and 15.1-5.

3.7 Main Steamline Break Radiological Analysis

A statement is added to Section 15.1.5 to state that a reduction in feedwater temperature to 420°F in Model E steam generators and the replacement of the Model E steam generators by Δ94 steam generators, operating at a feedwater temperature as low as 390°F, were evaluated.

A discussion of the revised methodology is provided. Tables 15.1-2 and 15.1-3 are revised to reflect a bounding analysis for all three steam generator configurations (e.g. the current Model E steam generators at a nominal feedwater temperature of 440°F, the Model E steam generators at a reduced feedwater temperature of 420°F, and the Δ94 steam generators at a feedwater temperature as low as 390°F) and the fact that the fuel does not experience departure from nucleate boiling (DNB) during a main steamline break (MSLB). UFSAR Section 15.1.5.3 and Tables 15.1-2 and 15.1-3 are revised to delete the current analysis assuming 5% failed fuel (FF).

3.8 Voltage-Based Tube Repair Criteria Radiological lysis

New Section 15.1.6 and new Tables 15.1-4 and 15.1-5 are added to provide a discussion of the Voltage-Based Steam Generator Tube Repair Criteria Radiological Analysis.

3 9 Reactor Coolant Pump Shaft Seizure (Locked Rotor) Radiological Analysis

A statement is added to Section 15.3.3.3 to state that the locked rotor analysis presented in Tables 15.3-3 and 15.3-4 bounds:

- Model E steam generators at a nominal feedwater temperature f 440°F;
- Model E steam generators at a reduced feedwater temperature of 420°F; and
- Δ94 steam generators (at a feedwater temperature as low as 390°F).

The values for the Low Population Zone (LPZ) thyroid doses which appear in the current UFSAR Table 15.3-4 are in error.

The current analysis described in the UFSAR (Model E steam generators at a nominal feedwater temperature of 440°F), both the reduced feedwater case (Model E steam generators at a reduced feedwater temperature of 420°F), and the Δ94 steam generators (at a feedwater temperature as low as 390°F) yield very similar results for the whole body gamma dose and beta skin dose. The current analysis for the Model E steam generators at a feedwater temperature of 440°F yields the bounding thyroid dose. UFSAR Table 15.3-4 will be updated to reflect the bounding whole body gamma and beta skin doses and the correct thyroid doses.

3.10 RCCA Ejection Accident Radiological Analysis

A statement is added to Section 15.4.8.3.2 to state that the analysis presented in Tables 15.4-4 and 15.4-5 bounds the Model E steam generators at a feedwater temperature range of 420°F to 440°F and the Δ94 steam generators with feedwater temperatures as low as 390°F. In addition, UFSAR Tables 15.4-4 and 15.4-5 are updated to reflect the bounding analysis.

3.11 Steam Generator Tube Rupture Radiological Analysis

A statement is added to Section 15.6.3.3 denoting that the current analysis for Model E steam generators with a nominal feedwater temperature of 440°F, presented in Tables 15.6-3 and 15.6-4, bounds both the reduced feedwater case (Model E steam generators at a reduced feedwater temperature of 420°F) and the Δ94 steam generators with feedwater temperatures as low as 390°F. A clarification with regards to the steam release from the ruptured steam generator is also made. Accordingly, no changes are proposed to UFSAR Tables 15.6-3 or 15.6-4 due to either the reduction in feedwater temperature or the Δ94 steam generators.

3.12 LOCA Radiological Analysis

A statement is added to Section 15.6.5.3 to state that the current analysis for Model E steam generators with a nominal feedwater temperature of 440°F bounds both the reduced feedwater case (Model E steam generators at a reduced feedwater temperature of 420°F) and the Δ94 steam generators with feedwater temperatures as low as 390°F.

3.13 Gaseous Radwaste Processing System Tank Failure Radiological Analysis

A statement is added to Section 15.7.1.3 to state that a reduction in feedwater temperature to 420°F in Model E steam generators or a replacement of the Model E steam generators with Δ94 steam generators, operating at a feedwater temperature as low as 390°F, will have a negligible impact on the isotopic inventory of the gaseous waste processing system and the radiological consequences of a GWPS failure.

3.14 Liquid Radwaste Processing System Tank Failure Radiological Analysis

A statement is added to Sections 15.7.2.3, and 15.7.3.3 to state that a reduction in feedwater temperature to 420°F in Model E steam generators or a replacement of the Model E steam generators by Δ94 steam generators, operating at a feedwater temperature as low as 390°F, will have a negligible impact on the isotopic inventory of the liquid waste processing system and the radiological consequences of a LWPS failure.

3.15 Impact of Operating at a Reduced Feedwater Temperature on Source Terms

New Section 15.A.6 is added to describe the impact of the reduction in feedwater temperature to 420°F in Model E steam generators on the fission product inventories in the fluid systems. The impact of the change on the reactor coolant inventory and the inventory in the secondary side was evaluated and determined to be negligible.

3.16 Impact of Δ94 Replacement Steam Generators on Source Terms

New Section 15.A.7 is added to describe the impact of the replacement of Model E steam generators with Δ94 steam generators, operating at a feedwater temperature as low as 390°F, on the fission product inventories in the fluid systems. The impact of the change on the reactor coolant inventory and the inventory in the secondary side was evaluated and determined to have a negligible impact on the activities of these systems.

3.17 General Accident Parameters

A statement is added referring the reader to a new Section 15.B.4 for a discussion of dose conversion factors.

3.18 Dose Conversion Factors

New Section 15.B.4 is added to describe the usage of dose conversion factors based upon either Regulatory Guide 1.109 or the International Commission on Radiation Protection Report 30 (ICRP 30). The references for section for 15.B is updated. The ICRP 30-based dose conversion factors are added to Table 15.B-3.

4. SAFETY EVALUATION

4.1 Introduction

Westinghouse and the South Texas Project Nuclear Operating Company (STPNOC) have performed several analyses of the proposed changes. The analyses considered:

- the current Model E steam generators operating at a nominal feedwater temperature of 440°F;
- the Model E steam generators operating at a reduced feedwater temperature of 420°F; and
- the replacement Δ94 steam generators operating at a feedwater temperature as low as 390°F.

The limiting parameters from the above three cases were used to create a bounding analysis. The following evaluation is a summary of these analyses.

The following UFSAR sections and accidents were evaluated for impact and additional analyses were performed, as necessary:

- Primary Side Isotopic Analysis (Section 11.1)
- Secondary Side Isotopic Analysis (Section 11.1)
- Main Steamline Break (MSLB) (Section 15.1.5)
- Voltage-Based Repair Criteria Dose Analysis (new Section 15.1.6)
- Feedwater System Pipe Break (Section 15.2.8)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor) (Section 15.3.3)
- Reactor Coolant Pump Shaft Break (Section 15.3.4)
- Control Rod Ejection Accidents (Section 15.4.8)
- Small Line Break Outside Containment (Section 15.6.2)
- Steam Generator Tube Rupture (SGTR) (Section 15.6.3)
- Loss of Coolant Accident (LOCA) (Section 15.6.5)
- Gaseous Radwaste Processing System Tank Failure (Sections 11.3 and 15.7.1)
- Liquid Radwaste Processing System Tank Failure (Sections 11.2 and 15.7.2)
- ICRP 30-based Dose Conversion Factors (new Section 15.B.4)

The effects on the system isotopic content (UFSAR Section 11.1) were determined to be negligible and the effects on the Chapter 15 offsite dose analyses are minimal. The results of the analyses are as follows:

4.2 Primary Side Isotopic Analysis

The analysis which determined the primary side isotopic content was reviewed for impact based on changes to the reactor coolant system (RCS) volume due to replacement of the Model E steam generators with the Δ94 steam generators. The change in RCS volume is very small and there is negligible impact on the isotopic content of the RCS. This analysis is described in UFSAR Section 11.1.

4.3 Secondary Side Isotopic Analysis

The analysis which determines the isotopic concentrations in the secondary side was reviewed for impact based on changes in the RCS isotopic inventory and secondary side volume change due to the Δ94 steam generators. The change in secondary side volume is slight and there is negligible impact on the isotopic content of the secondary side systems. This analysis is described in UFSAR Section 11.1.

4.4 Main Steamline Break (MSLB)

The effects of the Δ94 steam generators and the lower feedwater temperature on the MSLB were examined. Using parameters from the current Model E steam generators at feedwater temperatures of 440°F and 420°F, and the Δ94 steam generators, at a feedwater temperature as low as 390°F, a limiting analysis was performed.

A comparison of the parameters used in the analyses is presented in Table 4.4-1. The system parameters used for the revised analyses bound the three scenarios:

1. Current UFSAR Analysis: Model E SG at 440°F feedwater temperature
2. Reduced Feedwater Temperature: Model E SG at 420°F feedwater temperature
3. Δ94 Replacement Steam Generators at 390°F feedwater temperature

The values for steam releases used in the revised analysis bound the values for the current Model E steam generators (at 440°F feedwater temperature).

Table 4.4-1
Comparison of MSLB Analyses

Parameter	Current UFSAR Analysis (Model E SG @ 440°F FW)	Reduced FW Temperature (Model E SG @ 420°F FW)	Δ94 SGs (@ 390°F FW)	Value Used in Revised Analysis
Radiological Source Terms:				
Initial RCS Iodine				
Pre-existing I Spike:	60 µci/gm DEI (Tech Spec limit)	60 µci/gm DEI (Tech Spec limit)	60 µci/gm DEI (Tech Spec limit)	60 µci/gm DEI (Tech Spec limit)
Accident Spike:	1% FF + 500x escape rate	1% FF + 500x escape rate	1% FF + 500x escape rate	1% FF + 500x escape rate
Initial RCS Noble Gas Concentration	1% FF	1% FF	1% FF	1% FF
Initial Secondary-side Iodine Concentration				
Pre-existing I Spike:	0.1 µci/gm DEI (Tech Spec limit)	0.1 µci/gm DEI	0.1 µci/gm DEI	0.1 µci/gm DEI
Accident Spike:	1% FF with 1 gpm p/s leakage	0.1 µci/gm DEI	0.1 µci/gm DEI	0.1 µci/gm DEI
Initial Secondary-side Noble Gas Concentration	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage
Density of RCS (lbm/gal)	8.33 (room temp conditions)	8.33	8.33	8.33
System Masses/Volumes:				
RCS Mass (gm)	2.6E+8	2.6E+8	2.658E+8	2.6E+8 code input 2.658E+8 for determination of total RCS curie content
SG Mass (lbm)				
One SG	138,000	164,853	164,853	165,000
Four SGs	552,000	659,412	659,412	659,412
RCS Volume (ft^3)	13,103	13,103	13,521	13,521 (results in more total curies)

Table 4.4-1
Comparison of MSLB Analyses

Parameter	Current UFSAR Analysis (Model E SG @ 440°F FW)	Reduced FW Temperature (Model E SG @ 420°F FW)	Δ94 SGs (@ 390°F FW)	Value Used in Revised Analysis
Steam Releases (lbm):				
Intact Loops				
0-2 hr	431,000 (484,000 in analysis)	434,000	451,901	452,000
2-8 hr	1,068,000 (1,106,000 in analysis)	1,068,000	1,078,896	1,080,000
Faulted Loop				
0- 30 min.	210,000 (211,390 in analysis)	210,000	214,000	Note 1
MSIV Above Seat Drains Leak Rate				
Intact Loops				
0-36 hr	347.4 lbm/min	347.4 lbm/min	347.4 lbm/min	347.4 lbm/min
Faulted Loop	347.4 lbm/min (for first 8 hrs only)			0 lbm/min (All primary leakage is released directly from the break)

NOTE 1: The total curie inventory of the faulted steam generator is assumed to be instantaneously released at Time = 0 seconds. The primary side leakage into the faulted steam generator is instantaneously released to the environment and this release is assumed to continue for 36 hours.

The differences in the model parameters between the current Model E generators at 440°F and 420°F feedwater temperatures and the Δ94 steam generators at a feedwater temperature as low as 390°F are slight and the impact on offsite doses is small. A radiological analysis of the proposed changes was performed and the results are presented in Table 4.4-2.

Since the fuel does not experience DNB, a scenario considering fuel clad damage is not considered. This is in agreement with UFSAR Section 15.1.5.2, and with the NRC Safety Evaluation Report (SER) on voltage-based repair criteria (Reference 4). UFSAR Section

15.1.5.3 and UFSAR Tables 15.1-2 and 15.1-3 will be revised to delete the current analysis assuming 5% failed fuel.

The increase in dose consequences is due to a change in modeling methodology and not due to either the proposed reduction in feedwater temperature (for Model E steam generators) or the proposed Δ94 steam generators. The current analyses, as described in the UFSAR, assume the primary-to-secondary (p/s) leakage in the faulted steam generator is diluted in the water volume of the faulted steam generator (using the nominal at power water volume of the Model E steam generator). However, it is more conservative to assume the primary-to-secondary leakage in the faulted steam generator instantly flashes to steam and is released to the environment. Also, as per the Standard Review Plan (NUREG-0800), the initial iodine concentration in the secondary side was assumed to be at the Technical Specification limit of 0.1μci/gm dose equivalent Iodine-131 (DEI). Both analyses used an iodine partition factor of 1.0 for iodine released from the faulted steam generator. Therefore, the analysis for the current Model E steam generators at 440°F and 420°F feedwater temperatures, and the Δ94 steam generators at a feedwater temperature as low as 390°F, is more conservative than the current analysis described in the UFSAR.

Table 4.4-2
Comparisons of the MSLB Radiological Analyses

	Current UFSAR: Model E @ 440°F FW	Model E @ 420°F FW and Δ94 SG @ 390°F FW	Regulatory Limit
Pre-existing Iodine Spike			
EAB (rem)			
Thyroid	0.963	1.37	300
Whole Body	1.81E-3	2.78E-3	25
Beta-skin	6.5E-4	9.65E-4	25
LPZ (rem)			
Thyroid	0.769	0.88	300
Whole Body	1.14E-3	1.42E-3	25
Beta-skin	5.44E-4	6.28E-4	25

Table 4.4-2
Comparisons of the MSLB Radiological Analyses

	Current UFSAR: Model E @ 440°F FW	Model E @ 420°F FW and Δ94 SG @ 390°F FW	Regulatory Limit
Accident-induced Iodine Spike			
EAB (rem)			
Thyroid	1.81	4.12	30
Whole Body	5.32E-3	1.36E-2	2.5
Beta-skin	1.64E-3	3.91E-3	2.5
LPZ (rem)			
Thyroid	2.33	3.61	30
Whole Body	3.81E-3	7.50E-3	2.5
Beta-skin	1.5E-3	2.54E-3	2.5

4.5 Voltage-Based Repair Criteria Dose Analysis

The voltage-based repair criteria dose analysis description that is presented in this section has already been submitted for NRC staff review in support of the STP Nuclear Operating Company application for voltage-based repair criteria for STP Unit 2 (References 12, 13 and 14). NRC approval of that previously submitted application would be applicable to the voltage-based repair criteria analysis aspect of this application. In that sense, the voltage-based repair criteria dose analysis information included in this application is provided for completeness only.

As part of the voltage-based steam generator repair criteria (References 1, 2, 3 and 4), a maximum allowable post-MSLB primary-to-secondary leak rate was determined. This calculation has been re-performed and baselined to the MSLB accident described in Section 4.4, above. The current analyses, as described in the UFSAR, assume the primary-to-secondary (p/s) leakage in the faulted steam generator is diluted in the water volume of the faulted steam generator (using the nominal at-power water volume of the Model E steam generator). However, it is more conservative to assume the primary-to-secondary leakage in the faulted steam generator instantly flashes to steam and is released to the environment. Two additional methodology changes were necessary to obtain a reasonable allowable post-MSLB primary-to-secondary leakage limit. First, the current analysis, as well as the MSLB analysis, modeled the release of isotopes from the reactor coolant system (RCS) by integrating the release from 0 to 8 hours and instantaneously releasing this total amount of activity at time zero. The revised analysis, however, used the correct iodine release rate from the core into the RCS and assumed instantaneous mixing in the RCS. The correct primary-to-secondary leak rate was then used to release the isotopes directly to the environment.

The second major change is the use of Dose Conversion Factors (DCFs) based upon the International Commission on Radiation Protection Report 30 (ICRP 30). A discussion of the derivation of these DCFs is presented in Section 4.15, below. Previous analyses were based on Regulatory Guide 1.109 and ICRP 2, as described in UFSAR Section 15B.

Since the fuel does not experience DNB (per UFSAR Section 15.1.5), a scenario considering fuel clad damage is not considered. This is in agreement with UFSAR Section 15.1.5.2, and with the NRC SER on voltage-based repair criteria (Reference 4). Consequently, the limiting condition is now the offsite Emergency Planning Zone (EPZ) / Low Population Zone (LPZ) dose for the accident-induced iodine spike scenario. Consistent with the previous voltage-based repair criteria analysis, the updated analysis uses a diffuse source atmospheric dispersion factor (χ/Q). The proximity of the release point [the power-operated relief valves (PORVs) located in the isolation valve cubicle] to the control room air intake justifies the use of a diffuse source atmospheric dispersion factor for the determination of control room doses. The re-analysis results in an increase in the maximum allowable post-MSLB primary-to-secondary leakage from 5.0 gpm to 15.4 gpm.

A comparison of the previous analysis to the revised analysis is provided in Table 4.5-1.

Table 4.5-1
Comparison of the Voltage-Based Repair Criteria Analyses and the MSLB

Parameter	Current Voltage-Based Repair Criteria Analysis (Model E SG @ 440°F FW)	Revised MSLB Analysis (Table 4.4-1)	Value Used in Revised Voltage-Based Repair Criteria Analysis
Radiological Source Terms:			
Initial RCS Iodine			
Fuel w/Clad Failure (no Iodine Spike)	5%	N/A	
Pre-existing I Spike:	N/A	60 µci/gm DEI (Tech Spec limit)	N/A
Accident Spike:	N/A	1% FF + 500x escape rate	1% FF + 500x escape rate
Initial RCS Noble Gas Concentration	1% FF	1% FF	1% FF
Initial Secondary-side Iodine Concentration			

Table 4.5-1
Comparison of the Voltage-Based Repair Criteria Analyses and the MSLB

Parameter	Current Voltage-Based Repair Criteria Analysis (Model E SG @ 440°F FW)	Revised MSLB Analysis (Table 4.4-1)	Value Used in Revised Voltage-Based Repair Criteria Analysis
Fuel w/Clad Failure (no Iodine Spike)	5%	N/A	N/A
Pre-existing I Spike:	N/A	0.1 µci/gm DEI	N/A
Accident Spike:	N/A	0.1 µci/gm DEI	0.1 µci/gm DEI
Initial Secondary-side Noble Gas Concentration	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage
Density of RCS (lbm/gal)	8.33 (room temp conditions)	8.33	8.33
Iodine Treatment for primary-to secondary leakage			
faulted S/G	primary-to secondary leakage is diluted into the SG water volume and released with a partition factor of 1.	primary-to secondary leakage is released directly to the environment and released with a partition factor of 1.	primary-to secondary leakage is released directly to the environment and released with a partition factor of 1.
intact S/Gs	primary-to secondary leakage is diluted into the SG water volume and released with a partition factor of 100	primary-to secondary leakage is diluted into the SG water volume and released with a partition factor of 100	primary-to secondary leakage is diluted into the SG water volume and released with a partition factor of 100
Release rate of iodides from RCS, 0-8 hrs	total integrated amount released at t=0	total integrated amount released at t=0	Actual release rate used
Dose Conversion Factors	ICRP 2/ Reg Guide 1.109	ICRP 2/ Reg Guide 1.109	ICRP 30

Table 4.5-1
Comparison of the Voltage-Based Repair Criteria Analyses and the MSLB

Parameter	Current Voltage-Based Repair Criteria Analysis (Model E SG @ 440°F FW)	Revised MSLB Analysis (Table 4.4-1)	Value Used in Revised Voltage-Based Repair Criteria Analysis
Duration of primary-to-secondary leakage			
faulted S/G	0-8 hr	0-36 hr	0-8 hr
intact S/Gs	0-8 hr	0-36 hr	0-8 hr
System Masses/Volumes:			
RCS Mass (gm)	2.6E+8	2.6E+8 code input 2.658E+8 for determination of total RCS curie content	2.6E+8 code input 2.658E+8 for determination of total RCS curie content
SG Mass (lbm)			
One SG	138,000	165,000	138,000
Four SGs	552,000	659,412	552,000
RCS Volume (ft^3)	13,103	13,521 (results in more total curies)	13,521 (results in more total curies)
Steam Releases (lbm):			
Intact Loops			
0-2 hr	431,000 (484,000 in analysis)	452,000	484,000
2-8 hr	1,068,000 (1,106,000 in analysis)	1,080,000	1,106,000
Faulted Loop			
0- 30 min	210,000	214,000	210,000 (for release of pre-accident SG inventory. P/S leakage released to environment immediately)

Table 4.5-1
Comparison of the Voltage-Based Repair Criteria Analyses and the MSLB

Parameter	Current Voltage-Based Repair Criteria Analysis (Model E SG @ 440°F FW)	Revised MSLB Analysis (Table 4.4-1)	Value Used in Revised Voltage-Based Repair Criteria Analysis
MSIV Above Seat Drains Leak Rate			
Intact Loops			
0-36 hr	347.4 lbm/min (for 8 hr only)	347.4 lbm/min	347.4 lbm/min
Faulted Loop	115.8 lbm/min (for first 8 hrs only)	0 lbm/min (All primary leakage is released directly from the break)	0 lbm/min (All primary leakage is released directly from the break)

Dose results from the analyses are provided in Table 4.5-2.

Table 4.5-2
Comparison of the Voltage-Based Repair Criteria Analyses Dose Results (Rem)

	Current Voltage-Based Repair Criteria Analysis (Model E SG @ 440°F FW)	Regulatory Limit	Revised Voltage-Based Repair Criteria Analysis	Regulatory Limit
Allowed primary-to-secondary leakage in faulted steam generator	5.0 gpm total		15.4 gpm total	
Limiting Scenario	5% Failed Fuel (No Iodine Spike)		Accident-induced Iodine Spike	
0-2 hr EAB				
Thyroid (CDE[1])	133	300	15.0	30
Whole Body (DDE[2])	0.568	25	0.0503	2.5
Beta-skin (SDE[3])	0.192	25	0.0274	2.5
0-30 day LPZ				
Thyroid (CDE)	108	300	26.9	30
Whole Body (DDE)	0.285	25	0.0491	2.5
Beta-skin (SDE)	0.101	25	0.0282	2.5
30 day Control Room				
Thyroid (CDE)	19.5	30	5.55	30
Whole Body (DDE)	0.0935	5	0.00429	5
Beta-skin (SDE)	0.931	30	0.134	30
30 day TSC				
Thyroid (CDE)	27	30	7.69	30
Whole Body (DDE)	0.0596	5	0.00259	5
Beta-skin (SDE)	1.04	30	0.138	30

[1] Committed Dose Equivalent per ICRP 30
[2] Deep Dose Equivalent per ICRP 30
[3] Shallow Dose Equivalent per ICRP 30

4.6 Feedwater System Pipe Break

Per UFSAR Section 15.2.8.3, the feedwater line break with the most significant consequences occurs inside the containment between a steam generator and the feedwater check valve. In this case, the contents of the steam generator would be released to the containment. Since no fuel failures are postulated, the radioactivity released is less than that for the steam line break. Furthermore, automatic isolation of the containment would further reduce any radiological consequences from this postulated event. Therefore, the proposed changes do not impact this analysis.

4.7 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

The secondary iodine concentration is based upon an iodine concentration of 0.1 µci/gram of secondary coolant. A comparison of the steam released from this accident for the current and proposed change is given in Table 4.7-1.

Table 4.7-1
Comparison of Reactor Coolant Pump Shaft Seizure (Locked Rotor) Analyses

Parameter	Current Analysis (Model E SG @ 440°F FW)	Reduced FW Temperature (Model E SG @ 420°F FW)	Δ94 SGs (@ 390°F FW)	Value Used in Reduced FW Temperature (Model E SG @ 420°F FW)/Δ94 Analysis
Radiological Source Terms:				
Initial RCS Iodine				
Pre-existing I Spike:	60 µci/gm DEI (Tech Spec limit)	60 µci/gm DEI	60 µci/gm DEI	60 µci/gm DEI
Initial RCS Noble Gas Concentration	1% FF	1% FF	1% FF	1% FF
Density of RCS (lbm/gal)	8.33 (room temp conditions)	8.331	8.33	8.33
Initial Secondary-side Iodine Concentration				
Pre-existing I Spike:	0.1 µci/gm DEI (Tech Spec limit)	0.1 µci/gm DEI	0.1 µci/gm DEI	0.1 µci/gm DEI
Initial Secondary-side Noble Gas Concentration	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage

Table 4.7-1
Comparison of Reactor Coolant Pump Shaft Seizure (Locked Rotor) Analyses

Parameter	Current Analysis (Model E SG @ 440°F FW)	Reduced FW Temperature (Model E SG @ 420°F FW)	Δ94 SGs (@ 390°F FW)	Value Used in Reduced FW Temperature (Model E SG @ 420°F FW)/Δ94 Analysis
System Masses/Volumes:				
RCS Mass (gm)	2.6E+8	2.6E+8	2.658E+8	2.6E+8 for releases 2.658E+8 for determination of total RCS curie content
RCS Volume (cu ft)	13,103	13,103	13,521	13,521 (results in more total curies)
SG Mass (lbm)				
One SG	138,000	164,853	164,853	164,853
Four SGs	552,000	659,412	659,412	659,412
Secondary Side Steam Releases (lbm):				
0-2 hr	614,000	430,185	455,047	455,047
2-8 hr	1,264,000	1,130,113	1,137,757	1,137,757
Steam Flow (lbm/hr):				
	16,858,312	16,400,000	15,740,000	15,740,000

Using parameters from the current Model E steam generators at feedwater temperatures of 420°F and the Δ94 steam generators, at a feedwater temperature as low as 390°F, a limiting analysis was performed and compared to the current analysis of the current Model E steam generators at a nominal feedwater temperature of 440°F. The results are presented in Table 4.7-2.

Table 4.7-2
Comparison of Reactor Coolant Pump Shaft Seizure (Locked Rotor) Radiological Analyses

	UFSAR Analysis (Model E SG @ 440°F FW)	Bounding Analysis for Model E SG @ 420°F FW and Δ94 SGs (@ 390°F FW)	Current UFSAR Table 15.3-4	Revised UFSAR Table 15.3-4	Regulatory Limit
	10% Failed Fuel				
EAB (rem)					
Thyroid	1.04	0.66	1.1	1.1	30
Whole Body	0.038	0.04	0.038	0.040	2.5
Beta-skin	0.021	0.02	0.021	0.021	2.5
LPZ (rem)					
Thyroid	1.53	1.12	1.4	1.6	30
Whole Body	0.022	0.03	0.022	0.030	2.5
Beta-skin	0.013	0.02	0.013	0.020	2.5
	15% Failed Fuel				
EAB (rem)					
Thyroid	1.56	0.99	1.6	1.6	30
Whole Body	0.057	0.05	0.057	0.057	2.5
Beta-skin	0.031	0.03	0.031	0.031	2.5
LPZ (rem)					
Thyroid	2.28	1.67	2.1	2.3	30
Whole Body	0.034	0.05	0.034	0.050	2.5
Beta-skin	0.019	0.03	0.019	0.030	2.5

The Low Population Zone (LPZ) thyroid doses which appear in the current UFSAR Table 15.3-4 are in error and should reflect the values presented above for the Model E steam generators at a feedwater temperature of 440°F.

Note that the current analysis presented in the UFSAR (Model E steam generators at a nominal feedwater temperature of 440°F), both the reduced feedwater case (Model E steam generators at a reduced feedwater temperature of 420°F), and the Δ94 steam generators (at a feedwater temperature as low as 390°F) yield very similar results for the whole body gamma dose and beta skin dose. The current analysis for the Model E steam generators at a feedwater temperature of 440°F yields the bounding thyroid dose. UFSAR Table 15.3-4 will be updated to reflect the bounding whole body gamma and beta skin doses.

4.8 Reactor Coolant Pump Shaft Break

The radiological consequences of this accident are bounded by the RCP locked rotor accident discussed above; therefore, no specific analysis was performed for this accident. Neither the reduction of feedwater temperature nor the proposed replacement of the Model E steam generators by Δ94s has an effect on this conclusion.

4.9 Control Rod Ejection Accidents

The control rod ejection dose analysis was revised to reflect the reduced feedwater temperature case for the Model E steam generators and the Δ94 steam generators. These parameters are given in Table 4.9-1.

Table 4.9-1
Comparison of Control Rod Ejection Analysis Results

Parameter	Current Analysis (Model E SG @ 440°F FW)	Reduced FW Temperature (Model E SG @ 420°F FW)	Δ94 SGs (@ 390°F FW)	Value Used in Bounding Analysis
Radiological Source Terms:				
Power (MWt)	4,100	4100	4100	4100
Containment Spray	No	No	No	No
Containment Volume				
dilution (ft^3):	3.2E+6	3.38E+6	3.38E+6	3.38E+6
leakage (ft^3):	3.41E+6	3.41E+6	3.41E+6	3.41E+6
Density of RCS (lbm/gal)	8.33 (room temp conditions)	8.33	8.33	8.33
Activity Released from the Fuel:				
% Fuel Melt	0.25	0.25	0.25	0.25
Pellet Activity Released from melted fuel pins to RCS	50% of Iodides 100% of Noble Gases	50% of Iodides 100% of Noble Gases	50% of Iodides 100% of Noble Gases	50% of Iodides 100% of Noble Gases
Pellet Activity Released from melted fuel pins to RCB	25% of Iodides 100% of Noble Gases	25% of Iodides 100% of Noble Gases	25% of Iodides 100% of Noble Gases	25% of Iodides 100% of Noble Gases
% Clad Failure	10	10	10	10
Gap Activity Released from clad-damaged fuel pins to RCS	100% of iodides 100% of noble gases	100% of iodides 100% of noble gases	100% of iodides 100% of noble gases	100% of iodides 100% of noble gases

Table 4.9-1
Comparison of Control Rod Ejection Analysis Results

Parameter	Current Analysis (Model E SG @ 440°F FW)	Reduced FW Temperature (Model E SG @ 420°F FW)	Δ94 SGs (@ 390°F FW)	Value Used in Bounding Analysis
Gap Activity Released from clad-damaged fuel pins to RCB	100% of iodides 100% of noble gases	100% of iodides 100% of noble gases	100% of iodides 100% of noble gases	100% of iodides 100% of noble gases
Initial Fluid System Source Terms:				
Initial RCS Iodine				
Pre-existing I Spike:	60 μci/gm DEI (Tech Spec limit)	60 μci/gm DEI	60 μci/gm DEI	60 μci/gm DEI
Initial RCS Noble Gas Concentration	1% FF	1% FF	1% FF	1% FF
Initial Secondary-side Iodine Concentration				
Pre-existing I Spike:	0.1 μci/gm DEI (Tech Spec limit)	0.1 μci/gm DEI	0.1 μci/gm DEI	0.1 μci/gm DEI
Initial Secondary-side Noble Gas Concentration	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage	1% FF with 1 gpm p/s leakage
Iodine Partition Factor in SGs	0.01	0.01	0.01	0.01
System Masses/Volumes:				
RCS Mass (gm)	2.6E+8	2.6E+8	2.6E+8	2.6E+8
SG Mass (lbm)	552,000	659,000	659,000	659,000
Release Timing:				
Max. time to reach equilibrium between primary and secondary (sec.)	1250	4500	4500	4500
Min. time to release initial secondary side mass (sec.)	300	191	191	191
Steam Releases:				
Release of secondary mass (lbm)	72,300	1.56E+7	1.56E+7	1.56E+7
Minimum Secondary steam flow rate (lbm/hr)	16.86E+6 to 16.96E+6	16.40E+6 to 16.50E+6	15.74E+6 to 15.82E+6	15.74E+6

The results of the analysis show that the proposed changes do not result in a significant increase in consequences. Table 4.9-2 illustrates the large margin from the current doses to the acceptance criteria. UFSAR Tables 15.4-4 and 15.4-5 are to be updated to reflect the bounding analysis.

Table 4.9-2
Comparison of Control Rod Ejection Analyses

	UFSAR Table 15.4-5			Bounding Analysis			
	RCB Leakage	Secondary Systems	Total	RCB Leakage	Secondary Systems	Total	Dose Limit
EAB (rem)							
Thyroid	35.7	1.0	36.7	28.8	1.3	30	75
Whole Body	0.12	0.0051	0.125	0.1	0.5	0.6	6
Beta-skin	0.040	0.0017	0.0417	0.04	0.2	0.2	6
LPZ (rem)							
Thyroid	48.9	0.29	49.2	37.4	0.4	38	75
Whole Body	0.076	0.00149	0.0775	0.07	0.2	0.3	6
Beta-skin	0.029	0.00051	0.0295	0.03	0.05	0.1	6

The decrease in the thyroid dose due to the primary side releases is due to the correction of an error in the current analysis which places too much Iodine-131 in the clad gap. UFSAR Table 15.4-5 has been updated to reflect the bounding analysis in Table 4.9-2, above. The increase in the dose from the secondary side is the result of both a decrease in the time needed to release the contents of a steam generator (300 seconds to 191 seconds) and the longer time needed to equalize pressure between the primary and secondary sides (1250 seconds to 4500 seconds). A total of 1.56×10^7 pounds of steam is conservatively assumed to be discharged from the secondary system through the safety valves for 4500 seconds following the accident. Steam release is terminated after this time. The minimum time to release the initial steam generator mass is 191 seconds. The release rate necessary to release the total steam generator mass of 659,000 pounds in 191 seconds is 207,000 lbm/min. Assuming this flow rate is constant for 4500 seconds yields a total mass release of 1.56×10^7 pounds. This is the same methodology that was used in the current analysis.

4.10 Steam Generator Tube Rupture (SGTR)

The major impact of the feedwater temperature reduction on the SGTR is an increase in the initial mass in the steam generators which will increase the final water volume in the ruptured steam generator. An increase in the initial steam generator mass will lower the offsite radiological consequences since it ultimately leads to a slightly reduced steam rate from the steam generators. The steam releases for the reduced feedwater temperature cases for the Model E steam generators are bounded by those currently assumed in the UFSAR Table 15.6-3. Therefore, the proposed reduction of feedwater temperature in the Model E steam generators will result in lower radiological consequences.

A separate analysis was performed for the Δ94 steam generators. A comparison of the parameters used in the analyses is presented in Table 4.10-1.

Table 4.10-1
Comparison of Steam Generator Tube Rupture Analyses

Parameter	Current Analysis (Model E SG @ 440°F FW)	Value Used in Revised Analysis for Δ94 SG (@ 390°F FW)
Radiological Source Terms:		
Power (MWt)	3800	3800
Initial RCS Iodine		
Pre-existing I Spike:	60 μci/gm DEI (Tech Spec limit)	60 μci/gm DEI
Accident-initiated I Spike:	1.0 μci/gm DEI (Tech Spec limit) + 500x escape rate coefficient	1.0 μci/gm DEI (Tech Spec limit) + 500x escape rate coefficient
Rupture Flow Iodine Partition Factor	for fraction of flow that flashes in the SG, PF=1.0	for fraction of flow that flashes in the SG, PF=1.0
Initial RCS Noble Gas Concentration	1% FF	1% FF
Density of RCS leakage into intact SGs (lbm/gal)	5.9 (RCS conditions)	8.34 (cold conditions)
Primary to Secondary Leakage (gpm)		
Pre-accident	1	1
Post-accident:		
Intact SGs	0.233 / SG	0.333 / SG
Ruptured SG	0.30	0.0
Initial Secondary-side Iodine Concentration		
Pre-existing I Spike:	0.1 μci/gm DEI (Tech Spec limit)	0.1 μci/gm DEI
Accident-initiated I Spike:	0.1 μci/gm DEI (Tech Spec limit)	0.1 μci/gm DEI
Initial Secondary-side Noble Gas Concentration	0	0

Table 4.10-1
Comparison of Steam Generator Tube Rupture Analyses

Parameter	Current Analysis (Model E SG @ 440°F FW)	Value Used in Revised Analysis for Δ94 SG (@ 390°F FW)
Iodine Partition Factor in SGs	0.01	0.01
Time to Isolate Ruptured SG[1]	25 min	25 min
System Masses/Volumes:		
SG Mass (lbm)		
Ruptured SG	148,962	145,942
Steam Releases (lbm):		
Intact Loops		
0-2 hr	640,400	633,300
2-8 hr	1,051,100	1,322,600
Ruptured Loop		
0-2 hr	152,300	213,000
2-8 hr	41,700	35,200
RCS released to the ruptured SG	186,000	136,100
MSIV Above Seat Drains:		
Intact Loops		
0-36 hr	total of 347.4 lbm/min	total of 347.4 lbm/min
Ruptured Loop		
0-36 hr	total of 115.8 lbm/min	total of 115.8 lbm/min

Table 4.10-2 provides the results of the radiological analyses.

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[1] Includes 10 minutes for the operators to identify the ruptured steam generator and attempt to close the power operated relief valve (PORV) on the ruptured steam generator and 15 minutes to manually close the PORV block valve on the failed open PORV.

Table 4.10-2
Comparison of Steam Generator Tube Rupture Radiological Analyses

	Current UFSAR: Model E SG @ 440°F FW	Δ94 SG @ 390°F FW	Regulatory Limit
	Pre-Existing Iodine Spike		
EAB (rem)			
Thyroid	18.8	17.1	300
Whole Body	0.077	0.061	25
Beta-skin	0.12	0.088	25
LPZ (rem)			
Thyroid	8.4	5.1	300
Whole Body	0.028	0.018	25
Beta-skin	0.053	0.026	25
	Accident-induced Iodine Spike		
EAB (rem)			
Thyroid	7.4	4.6	30
Whole Body	0.085	0.055	2.5
Beta-skin	0.123	0.085	2.5
LPZ (rem)			
Thyroid	5.2	1.4	30
Whole Body	0.033	0.016	2.5
Beta-skin	0.053	0.025	2.5

The current SGTR analysis for the Model E steam generators with a nominal feedwater temperature of 440°F remains bounding. There are no changes to UFSAR Tables 15.6-3 and 15.6-4 due to the proposed changes.

4.11 Small Line Break Outside Containment

Neither the reduction of feedwater temperature nor the replacement of the Model E steam generators with Δ94 steam generators impacts the core isotopic inventory or the assumptions used to analyze the radiological consequences of a Small Line Break Outside Containment. Both a letdown line break and a sample line break were previously analyzed. These analyses were evaluated based on changes in the RCS isotopic inventory. Since the change in RCS isotopic inventory was negligible, the impact on the current analyses is negligible and the results remain valid and within the Standard Review Plan (NUREG-0800) acceptance criteria of a "small fraction" of the 10 CFR 100 limits. A "small fraction" of 10 CFR 100 means 10 percent of these exposure guideline values (i.e., 2.5 rem and 30 rem for the whole-body and thyroid doses, respectively).

4.12 Loss of Coolant Accident (LOCA)

Neither the reduction of feedwater temperature nor the replacement of the Model E steam generators with Δ94 steam generators impacts the core isotopic inventory or the assumptions used to analyze the radiological consequences of a postulated LOCA. The primary effect on the LOCA analysis is the amount of RCS inventory released to the environment via the supplemental purge valve. A review of the parameters involved indicated that the current analysis is bounding. Therefore, the LOCA radiological analysis is not impacted by this change.

4.13 Gaseous Radwaste Processing System Tank Failure

This analysis examines the radiological consequences of a failure of a Gaseous Radwaste Processing System tank. The GWPS Tank Failure radiological calculation was evaluated based on changes in the RCS isotopic inventory (which would cause changes in the system inventory of the GWPS tanks). Since the change in RCS isotopic inventory was negligible, the impact on the current analysis is negligible and the results remain valid and acceptable.

4.14 Liquid Radwaste Processing System Tank Failure

This analysis examines the radiological consequences of a failure of a Liquid Radwaste Processing System tank. The LWPS Tank Failure radiological calculation was evaluated based on changes in the RCS isotopic inventory (which would cause changes in the system inventory of the LWPS [illegible]). Since the change in RCS isotopic inventory is negligible, the impact on the current an[illegible] is negligible and the results remain valid and within the Standard Review Plan (NUREG-0800) acceptance criterion of not exceeding the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest drinking water source.

4.15 ICRP 30-based Dose Conversion Factors

The radiological analyses presented in the STP UFSAR are based upon ICRP 2 and Regulatory Guide 1.109 (References 5 and 7) dose conversion factors (DCFs). However, to increase the allowable primary-to-secondary leakage for the voltage-based repair criteria analysis presented in Section 4.5, above, it was necessary to use DCFs based upon ICRP 30 (Reference 8) and US EPA Federal Guidance Reports 11 and 12 (References 9 and 10). This section provides a discussion on the derivation of the DCFs used in the analysis from the data available in the

ICRP 30 and Federal Guidance Reports. Both the Regulatory Guide 1.109-based DCFs and the ICRP 30-based DCFs are presented in Table 4.15-1.

For certain analyses, dose conversion factors were derived from ICRP 30 data (Reference 8) as an alternative to those based on Regulatory Guide 1.109. These DCFs may be used as a replacement for the DCFs based upon Regulatory Guide 1.109 for control room, Technical Support Center (TSC), and offsite calculations. However, unless stated in the accident description, the DCFs based upon Regulatory Guide 1.109 were used in an analysis.

Thyroid DCF

The tabulated ICRP 30-based thyroid DCFs listed in Table 4.15-1 all originate from Federal Guidance Report 11 (Reference 9). These coefficients give committed dose equivalence (CDE) to the thyroid per unit activity of inhaled radionuclides. The coefficients were calculated using the most recent metabolic and physiologic modeling and should provide the best estimate of thyroid dose.

Skin DCF

The most recent publication for skin dose conversion factors is Federal Guidance Report 12. However, these reported DCF contain contributions to skin dose from both photons and electrons. The skin DCFs are partially corrected for gamma contribution based on the control room volume. This gives a more conservative dose calculation than beta alone. The total skin DCFs were taken from Reference 10, with the exception of Kr-89 and Xe-137, which were taken from Reference 11.

The larger volume of the control room will also make a conservative gamma correction to the skin DCF for use with the smaller Technical Support Center. This is because the Murphy-Campe (Reference 6) geometry factor term is inversely proportional to the volume, and the DCF correction is inversely related to the geometry factor, which makes the DCF directly related to the node volume. Therefore, the larger control room volume makes a conservatively larger DCF.

The skin DCFs are conservative to use for offsite doses. This is because the Regulatory Guide 1.109 for skin doses are based on beta exposure only. Including the control room volume-corrected gamma contribution in the offsite skin doses is more conservative than beta only.

Total Body DCF

The Total Body DCF taken from Federal Guidance Report 12 (Reference 10) assumes submersion in a semi-infinite cloud of effluent. The cloud concentration is assumed to be uniform throughout the problem domain. Whole body DCFs were taken from Reference 10, with the exception of Kr-89 and Xe-137, which were taken from Reference 11.

Table 4.15-1

Dose Conversion Factors

	ICRP 2 and Reg Guide 1.109 Based			ICRP 30 - Based		
	Total Body	Beta Skin	Thyroid	Total Body	Beta Skin	Thyroid
Nuclide	(rem-m^3/ ci-sec)	(rem-m^3/ ci-sec)	(rem/ci)	(rem-m^3/ ci-sec)	(rem-m^3/ ci-sec)	(rem/ci)
I-131	8.72E-2	3.17E-2	1.49E+6	6.734E-2	4.087E-2	1.080E+6
I-132	5.13E-1	1.32E-1	1.43E+4	4.144E-1	1.617E-1	6.438E+3
I-133	1.55E-1	7.35E-2	2.69E+5	1.088E-1	1.032E-1	1.798E+5
I-134	5.32E-1	9.23E-2	3.73E+3	4.810E-1	2.011E-1	1.066E+3
I-135	4.21E-1	1.29E-1	5.60E+4	2.953E-1	1.153E-1	3.130E+4
Kr-83M	2.40E-6	NA	NA	5.550E-6	1.547E-5	NA
Kr-85M	3.71E-2	4.63E-2	NA	2.768E-2	5.468E-2	NA
Kr-85	5.1E-4	4.25E-2	NA	4.403E-4	4.843E-2	NA
Kr-87	1.88E-1	3.08E-1	NA	1.524E-1	3.482E-1	NA
Kr-88	4.66E-1	7.51E-2	NA	3.774E-1	1.221E-1	NA
Kr-89	5.26E-1	3.2E-1	NA	3.232E-1	3.981E-1	NA
Xe-131m	2.9E-3	1.51E-2	NA	1.439E-3	1.544E-2	NA
Xe-133m	7.96E-3	3.15E-2	NA	5.069E-3	3.227E-2	NA
Xe-133	9.32E-3	9.70E-3	NA	5.772E-3	1.145E-2	NA
Xe-135m	9.89E-2	2.25E-2	NA	7.548E-2	3.144E-2	NA
Xe-135	5.38E-2	5.90E-2	NA	4.403E-2	7.066E-2	NA
Xe-137	4.50E-2	3.87E-1	NA	3.026E-2	4.642E-1	NA
Xe-138	2.80E-1	1.31E-1	NA	2.135E-1	1.728E-1	NA

5. DATA FOR CONFIRMATORY DOSE ANALYSES

The South Texas Project provides the following information to support the confirmatory dose analyses. This section provides the input parameters and major assumptions used in the design dose analyses discussed in Section 4.

5.1 Main Steam Line Break (MSLB)

The parameters used in the MSLB analysis for the Model E Reduced Feedwater Temperature / Δ94 Steam Generator are presented in Table 5.1-1, below. Resultant doses are presented in Table 4.4-2.

Table 5.1-1
MSLB Analysis Parameters

Parameter		Value
Flashing fraction for primary-to-secondary leakage into intact SGs		Not used, see assumptions
Scrubbing fraction for flashed portion of primary-to-secondary leakage into the intact SGs.		Not used, see assumptions
Primary bypass fraction (liquid entrained in the flashing fraction) for intact SGs		Not used, see assumptions
Time to isolate faulted SG		30 minutes
Duration of plant cooldown by secondary side		8 hours
Primary coolant concentration for Technical Specification limit of 60 μci/gm DE^{131}I:		
Pre-existing Spike Values	I-131	45 μci/gm
(consistent with UFSAR Table 15.A-4)	I-132	53 μci/gm
	I-133	71 μci/gm
	I-134	11 μci/gm
	I-135	40 μci/gm
Primary Side Parameters:		
	Volume (ft^3)	13,521
	Pressure (psia)	2250
	Temperature (°F)	585.8 to 596.5
Secondary Side Parameters:		
	SG Mass (lbm)	
	One SG	165,000
	Four SGs	659,412
	Feedwater Temperature (°F)	390

Table 5.1-1
MSLB Analysis Parameters

Parameter	Value
Primary coolant DE^{131}I 48 hr Technical Specification limit	1 µci/gm
Primary coolant DE^{131}I Spike Technical Specification limit	60 µci/gm
Secondary coolant DE^{131}I Technical Specification limit	0.1 µci/gm
Primary-to-secondary leak rate (gpm):	
all SGs, total	1
Faulted SG	0.35
Intact SGs	0.65
Iodine Partition Factors:	
Faulted SG	1.0
Intact SGs	0.01
Steam Releases (lbm):	
Faulted SG	
0-30 min	214,000
Intact SGs	
0-2 hr.	452,000
2-8 hr.	1,080,000
MSIV Above Seat Drains on Intact SGs	
0-36 hr.	347.4 lbm/min
RCS Letdown Flow rate (gpm)	100
Release Rate for 48 hr Technical Specification limit of 1 µci/gm DE^{131}I:	
I-131	3.85E-3 ci/sec
I-132	5.60E-3 ci/sec
I-133	8.05E-3 ci/sec
I-134	8.75E-3 ci/sec
I-135	7.35E-3 ci/sec
Atmospheric Dispersion Factors:	
EAB: 0-2 hr	1.3E-4 sec/m^3
LPZ: 0-2 hr	3.8E-5 sec/m^3
LPZ: 2-8 hr	1.6E-5 sec/m^3

Since the LOCA is the limiting accident for the control room and technical support center doses, control room and technical support center doses were not calculated for this accident.

The conservative assumptions and parameters used to calculate the activity released and offsite doses for a steam line break are the following:

1. Prior to the accident, the secondary coolant specific activity is based upon equilibrium reactor coolant concentration with 1 gal/min primary-to-secondary leakage.

2. The accident does not result in failure of fuel rod cladding.

3. The primary-to-secondary leakage of 1 gal/min is assumed to continue for 36 hours following the accident. It is assumed that 0.35 gal/min leakage occurs in the defective SG and 0.217 gal/min in each of the unaffected SGs.

4. Offsite power is lost and main steam condensers are not available for steam dump operation.

5. Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. The only steam release after eight hours is through the main steam isolation valve (MSIV) above seat drain line flow restriction orifices.

6. The iodine partition factor in the steam generators is the ratio of the amount of iodine per unit mass of steam to the amount of iodine per unit mass of liquid and is equal to 0.01.

7. For a pre-existing iodine spike, the primary coolant concentrations are assumed to be equal to the Technical Specification limit for full power operation following an iodine spike. The secondary coolant specific activity is equal to the Technical Specification limit of 0.1 ci/gm dose equivalent I-131.

8. For an accident-initiated iodine spike, the primary coolant iodine concentrations are assumed to be functions of time. The spike is accounted for by increasing the source term or release rate from the fuel by a factor of 500. Prior to the accident, the secondary coolant concentration is based upon equilibrium reactor coolant concentration with 1 gal/min primary-to-secondary leakage.

9. The density of the RCS is assumed to be 8.33 lbm/ft^3 (room temperature conditions).

10. The primary-to-secondary leakage into the intact steam generators is assumed to be 0.65 gpm (at 8.33 lbm/ft^3). This activity is mixed in the intact steam generators. The release to the environment is modeled as a release of steam generator mass, passing through a "pseudo-filter" to model the 0.01 iodine partition factor.

5.2 Voltage-Based Repair Criteria Dose Analysis

The parameters used in this analysis are presented in Table 5.2-1. Resultant doses are presented in Table 4.5-2.

Table 5.2-1
Voltage-Based Repair Criteria Dose Analysis Parameters

Parameter	Value
Flashing fraction for primary-to-secondary leakage into intact SGs	Not used, see assumptions
Scrubbing fraction for flashed portion of primary-to-secondary leakage into the intact SGs	Not used, see assumptions
Primary bypass fraction (liquid entrained in the flashing fraction) for intact SGs	Not used, see assumptions
Time to isolate faulted SG	30 minutes
Duration of plant cooldown by the secondary side	8 hours
Primary Side Parameters:	
Volume (ft^3)	13,521
Pressure (psia)	2250
Temperature (°F)	585.8 to 596.5
Secondary Side Parameters:	
SG Mass (lbm)	
One SG	138,000
Four SGs	552,000
Feedwater Temperature (°F)	390
Secondary coolant DE^{131}I Technical Specification limit	0.1 μci/gm
Primary-to-secondary leak rate (gpm)	
all SGs, total pre-accident:	1
post-accident:	0.42
Faulted SG (post-accident)	0.147 (35% of total)
Intact SGs (post-accident)	0.273 (65% of total)
Iodine Partition Factors	
Faulted SG	1.0
Intact SGs	0.01
Steam Releases (lbm)	
Faulted SG	
0-30 min	210,000
Intact SGs	
0-2 hr.	484,000

Table 5.2-1
Voltage-Based Repair Criteria Dose Analysis Parameters

Parameter	Value
2-8 hr.	1,106,000
MSIV Above Seat Drains on Intact SGs	
0-36 hr.	347.4 lbm/min
RCS Letdown Flow rate (gpm)	100
Release Rate for 48 hr Technical Specification limit of 1 µci/gm DE^{131}I:	
I-131	3.85E-3 ci/sec
I-132	5.60E-3 ci/sec
I-133	8.05E-3 ci/sec
I-134	8.75E-3 ci/sec
I-135	7.35E-3 ci/sec
Atmospheric Dispersion Factors	
EAB: 0-2 hr	1.3E-4 sec/m^3
LPZ: 0-2 hr	3.8E-5 sec/m^3
LPZ: 2-8 hr	1.6E-5 sec/m^3
LPZ: 8-24 hr	1.1E-5 sec/m^3
LPZ: 24-720 hr	4.3E-6 sec/m^3
Control Room & TSC: 0-8 hr	1.06E-3 sec/m^3
Control Room & TSC: 8-24 hr	7.03E-4 sec/m^3
Control Room & TSC: 24-96 hr	4.45E-4 sec/m^3
Control Room & TSC: 96-720 hr	1.91E-4 sec/m^3
Control Room Parameters:	
Volume (ft^3)	274,080
Normal makeup flow (cfm/ train)	1000
Normal supply flow (cfm/ train)	17,400[1]
Normal return flow (cfm/ train)	16,400[1]
Emergency makeup flow (cfm/ train)	1100
Makeup Filter Efficiencies (2 of 3 HVAC trains operating)	
Elemental I	98.86%
Organic I	94.32%
Particulate I	99%
Unfiltered Inleakage	10 cfm
Recirculation Filter Flow Rate	4750 cfm / train

[1] This is a per train value, with multiple trains operating. Normally, only one train is in use. With one train normal operation, this flow may increase by as much as 20%.

Table 5.2-1
Voltage-Based Repair Criteria Dose Analysis Parameters

Parameter	Value
Recirculation Filter Efficiencies (2 of 3 HVAC trains operating)	
Elemental I	95%
Organic I	95%
Particulate I	99%
Number of HVAC trains operating (3 max):	
to determine maximum thyroid dose	2
to determine maximum whole body and beta-skin dose	3
Technical Support Center Parameters:	
Volume (ft^3)	48,170
Normal makeup flow (cfm)	1100
Normal supply flow (cfm)	11,600
Normal return flow (cfm)	10,500
Emergency makeup flow (cfm)	1210
Makeup Filter Efficiencies (%)	
Elemental I	99%
Organic I	99%
Particulate I	99%
Unfiltered Inleakage (cfm)	16.2
Recirculation Filter Flow Rate (cfm)	4750
Recirculation Filter Efficiencies (%):	
Elemental I	99%
Organic I	99%
Particulate I	99%
Occupancy Factors (Control room & TSC):	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4

The conservative assumptions and parameters used to calculate the activity released and offsite doses for a steam line break with additional primary-to-secondary leakage for the voltage-based repair criteria are the following:

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a 3 region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively.

2. The accident results in no failure of fuel rod cladding.

3. Reactor coolant density is 8.33 lbs/gal (room temperature conditions).

4. The equilibrium secondary activity before the Steam Generator rupture is based upon a preexisting primary to secondary leakage of 1 gpm. This is conservative since the Technical Specifications for the voltage-based repair criteria limits the preexisting leakage to 150 gpd per steam generator or 600 gpd (0.42 gpm) total.

5. The total steam generator tube leak rate prior to the accident and until 8 hours after the start of the accident is 0.42 gpm (approx. 600 gpd). This is conservatively divided into 0.147 gpm (35%) to the affected loop and 0.273 gpm (65%) to the unaffected loops.

6. For a pre-existing iodine spike, the activity in the reactor coolant is based upon an iodine spike which has raised the reactor coolant concentration to 60 micro Ci/gm of dose equivalent I-131. The secondary coolant activity is based on 0.1 micro Ci/gm of dose equivalent I-131. Noble gas activity is based on 1% failed fuel. This scenario was not analyzed since the pre-existing iodine case is less limiting than the accident-initiated iodine spike case.

7. For a accident-induced iodine spike, the accident initiates an iodine spike in the RCS which increases the iodine release rate from the fuel to a value 500 times greater than the release rate corresponding to a RCS concentration of 1 micro Ci/gm dose equivalent I-131. The iodine activity released from the fuel to the RCS is conservatively assumed to mix instantaneously and uniformly in the RCS. Noble gas activity is based on 1% failed fuel. Steam generator activities are assumed to be identical to the pre-existing iodine spike (iodides based on 0.1 micro Ci/gm of dose equivalent I-131; noble gas activity based on 1% failed fuel).

8. Following the accident, auxiliary feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. The primary-to-secondary leakage in the faulted steam generator is assumed to flash to steam and be immediately released to the atmosphere with an iodine partition factor of 1.0.

9. Following the accident, the primary to secondary leakage in the affected steam generator is assumed to be instantaneously released to the environment. This is conservative since the free volume above the top of tubes and the volume of the 30" pipe inside the containment provide space for accumulation and hold-up of radionuclides before release to the

environment. Another conservatism is that the plate-out effect was not modeled.

10. In the intact steam generators, the primary to secondary leakage is mixed into the existing mass in the steam generators. The release to the environment is modeled as a release of steam generator mass with an iodine partition factor of 0.01.

11. Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. This ceases the primary-to-secondary leakage. The only steam release after eight hours is the release of secondary side inventory through the MSIV above seat drain line flow restriction orifices. All releases stop 36 hours after the accident.

12. Offsite Power is lost; the condensers are unavailable for steam dump.

13. All activity is released to the environment with no consideration given to radioactive decay or to cloud depletion by ground deposition during transport to the exclusion zone boundary and low population zone.

14. The X/Q for the RCB to CR/TSC intake is assumed to apply for MSLB site to the CR/TSC intake.

15. The offsite, CR and TSC doses change linearly as a function of the primary to secondary break flow.

5.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

The parameters used in the Locked Rotor analysis for the Model E Reduced Feedwater Temperature / Δ94 Steam Generator are presented in Table 5.3-1, below. Resultant doses are presented in Table 4.7-2.

Table 5.3-1
Locked Rotor Analysis Parameters

Parameter	Value
Mass of liquid released from the SGs, as a function of time	0-2 hrs: 455,047 lbm 2-8 hrs: 1,137,757 lbm
Duration of plant cooldown by the secondary side (hr)	8
Fraction of fuel rods experiencing cladding perforation	10 % (a radiological analysis was also performed assuming a 15% rod failure)
Fraction of fuel rods experiencing melting	0 %

The primary-to-secondary leakage is assumed to continue at 1 gpm and thoroughly mix with the contents of each steam generator. The radiological release continues for 8 hours, via the steam releases from the steam generators. A partition factor of 0.01 is used for iodides in the steam generators.

Since the LOCA is the limiting accident for the control room and technical support center doses, control room and technical support center doses were not calculated for this accident.

The assumptions used to calculate the activity released and offsite doses for a pump shaft seizure accident are the following:

1. Prior to the accident, the primary coolant concentrations are assumed to be equal to the Technical Specification limit for full power operation following an iodine spike (I-131 dose equivalent of 60 μci/gm).

2. Prior to the accident, the secondary coolant specific activity is equal to the Technical Specification limit of 0.10 μci/gm dose equivalent

3. Ten percent of the total core fuel cladding is damaged, which results in the release to the reactor coolant of 10 percent of the total gap inventory of the core. A radiological analysis is also performed assuming a 15% rod failure. This activity is assumed to be uniformly mixed in the primary coolant.

4. The primary-to-secondary leakage of 1 gal/min (Technical Specification limit) is assumed to continue for 8 hours following the accident.

5. Offsite power is lost and main steam condensers are not available for steam dump operation.

6. Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. No further steam or activity is released to the environment.

7. The iodine partition factor in the steam generators is equal to 0.01.

5.4 Control Rod Ejection Analysis

The assumptions used in the control rod ejection analysis for the Model E Reduced Feedwater Temperature / Δ94 Steam Generator are discussed below. Values of major parameters are presented in Table 4.9-1. Resultant doses are presented in Table 4.9-2. Since the LOCA is the limiting accident for the control room and technical support center doses, control room and technical support center doses were not calculated for this accident.

Assumptions made for the Release via Containment Leakage Pathway:

1. One hundred percent of the noble gases and iodides in the clad gaps of the fuel rods experiencing clad damage (assumed to be 10 percent of the rods in the core) is assumed released to the containment.

2. The fraction of fuel melting was assumed to be 0.25 percent of the core as determined by the following method:

   a) A conservative upper limit of 50 percent of rods experiencing clad damage may experience centerline melting (a total of 5 percent of the core).

   b) Of the rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the volume actually melts (0.5 percent of the core could experience melting).

   c) A conservative maximum of 50 percent of the axial length of the rod would experience melting due to the power distribution (0.5 of the 0.5 percent of the core = 0.25 percent of the core).

3. Of the fuel melted, 100% of the noble gases and 25% of the iodides are assumed to be released to the containment and available for leakage to the atmosphere.

4. The clad gap activity is assumed to be 10% of the iodides, 30% of the Kr-85, and 10% of the noble gases.

5. The activity in the fuel pellet-clad gap and the activity released due to fuel melting is instantaneously mixed in the containment and available for release.

6. The containment leaks for the first 24 hours at its design leak rate of 0.3 percent per day. Thereafter, the containment leak rate is 0.15 percent per day.

7. The only removal processes considered for the containment are radioactive decay and leakage (i.e., no credit for containment sprays).

<u>Assumptions made for the Release via the Primary-to-Secondary Leakage Pathway:</u>
The model for the activity available for release to the atmosphere from the safety valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that activity leaking from the primary coolant through the steam generator tubes following the accident. The leakage of primary coolant to the secondary side of the steam generators is assumed to continue at its initial rate, assumed to be the same rate as the leakage prior to the accident, until the pressure in the primary and secondary systems are equalized. No mass transfer from the primary system to the secondary system is assumed thereafter. In the case of coincident loss of offsite power, activity is assumed to be released to the atmosphere through the steam generator safety valves.

The following assumptions were used in the analysis of the release of radioactivity to the environment in the event of a postulated rod ejection accident:

1. One hundred percent of the noble gases and iodides in the clad gaps of the fuel rods experiencing clad damage (assumed to be 10 percent of the rods in the core) are assumed to be released and instantaneously mixed into the reactor coolant system.

2. The fraction of fuel melting was assumed to be 0.25 percent of the core as determined by the following method:

   a) A conservative upper limit of 50 percent of rods experiencing clad damage may experience centerline melting (a total of 5 percent of the core).

   b) Of the rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the volume actually melts (0.5 percent of the core could experience melting).

   c) A conservative maximum of 50 percent of the axial length of the rod would experience melting due to the power distribution (0.5 of the 0.5 percent of the core equals 0.25 percent of the core).

3. Of the fuel melted, 100% of the noble gases and 50% of the iodides are assumed to be released and instantaneously mixed into the reactor coolant system.

4. The clad gap activity is assumed to be 10% of the iodides, 30% of the Kr-85, and 10% of the noble gases.

5. Primary and secondary system pressures are equalized after 4500 seconds, thus terminating primary-to-secondary leakage in the steam generators.

6. For the case of loss of offsite power, a total of 1.56×10^7 pounds of steam is discharged from the secondary system through the safety valves for 4500 seconds following the accident. Steam release is terminated after this time. The minimum time to release the initial steam generator mass is 191 seconds. The rate of release necessary to release the total steam generator mass of 659,000 pounds in 191 seconds is 207,000 lbm/min. Assuming this flow rate is constant for 4500 seconds yields a total mass release of 1.56×10^7 pounds.

5.5 Steam Generator Tube Rupture (SGTR) Accident

The parameters used in the SGTR analysis for the Model E Reduced Feedwater Temperature / Δ94 Steam Generator are presented in Table 5.5-1, below. Resultant doses are presented in Table 4.10-2.

Table 5.5-1
SGTR Analysis Parameters

Parameter	Value
Mass (lbm) released from ruptured SG, as a function of time	0-2 hrs: 213,000 2-8 hrs: 35,200
Mass (lbm) released from the intact SGs as a function of time.	0-2 hrs: 633,300 2-8 hrs: 1,322,600
Flashing fraction for primary-to-secondary leakage into SGs	ruptured SG: see figures of break flow and flashed break flow; intact SGs: 0.0
Ruptured Flow Iodine Partition Factor	for fraction of flow that flashes in the SG, PF = 1.0
Primary bypass fraction (liquid entrained in the flashing fraction) for the SGs	ruptured SG: 0.0 intact SGs: 0.0
Time to isolate ruptured SG[1]	25 minutes
Duration of plant cooldown by the secondary side	862 seconds (14.37 minutes)
Primary-to-secondary release rate from the ruptured tube as a function of time	see figure of break flow
Overfill Conditions exists?	No

[1] Includes 10 minutes for the operators to identify the ruptured steam generator and attempt to close the power operated relief valve (PORV) on the ruptured steam generator and 15 minutes to manually close the PORV block valve on the failed open PORV.

Table 5.5-1
SGTR Analysis Parameters

Parameter	Value
Primary coolant concentration for Technical Specification limit of 60 µci/gm DE^{131}I:	
I-131	46.14 µci/gm
I-132	53.82 µci/gm
I-133	73.08 µci/gm
I-134	10.98 µci/gm
I-135	40.38 µci/gm
Primary coolant activity due to a pre-existing I spike	
I-131	11676 ci
I-132	13614 ci
I-133	18492 ci
I-134	2778 ci
I-135	10218 ci
Primary coolant concentration due to an accident-initiated I spike	Accident initiated spike is assumed to continue until the primary coolant ^{131}I concentration reaches 11676 ci
Primary Side Parameters:	
RCS Volume (ft^3)	12512.4
RCS Initial Vessel Average Temperature (°F)	582.3
Initial Pressurizer Pressure (psia)	2204
Initial RCS Coolant Mass (lbm)	557,500
Secondary Side Parameters:	
Initial Mass (total)	512,000 lbm
Steam Volume (total)	30060 ft^3
Initial Steam Pressure (psia)	964
Feedwater Temperature (°F)	440
Primary coolant DE^{131}I 48 hr Technical Specification limit	60 µci/gm
Secondary coolant DE^{131}I Technical Specification limit	0.1 µci/gm
Primary-to-secondary leak rate:	
Ruptured SG	0 gpm
Intact SGs	1 gpm (total)
Iodine Partition Factors:	
Ruptured SG	0.01
Intact SGs	0.01

Table 5.5-1
SGTR Analysis Parameters

Parameter	Value
Steam Releases:	
Ruptured SG	
0-2 hr.	213,000 lbm
> 2 hr.	35,200 lbm
Intact SGs	
0-2 hr.	633,300 lbm
> 2 hr.	1,322,600 lbm
MSIV Above Seat Drains on Intact SGs	
0-36 hr.	347.4 lbm/min
RCS Letdown Flow rate	100 gpm
Release Rate for Technical Specification limit of 1.0 µci/gm DE^{131}I:	
I-131	43.63 ci/hr
I-132	87.0 ci/hr
I-133	34.98 ci/hr
I-134	40.38 ci/hr
I-135	31.56 ci/hr
500x Release Rate for Accident Initiated Spike:	
I-131	8160 ci/hr
I-132	43500 ci/hr
I-133	17490 ci/hr
I-134	20190 ci/hr
I-135	15780 ci/hr
Atmospheric Dispersion Factors:	
EAB: 0-2 hr	1.3E-4 sec/m^3
LPZ: 0-2 hr	3.8E-5 sec/m^3
LPZ: >2 hr	1.6E-5 sec/m^3

Figure 5.5-1

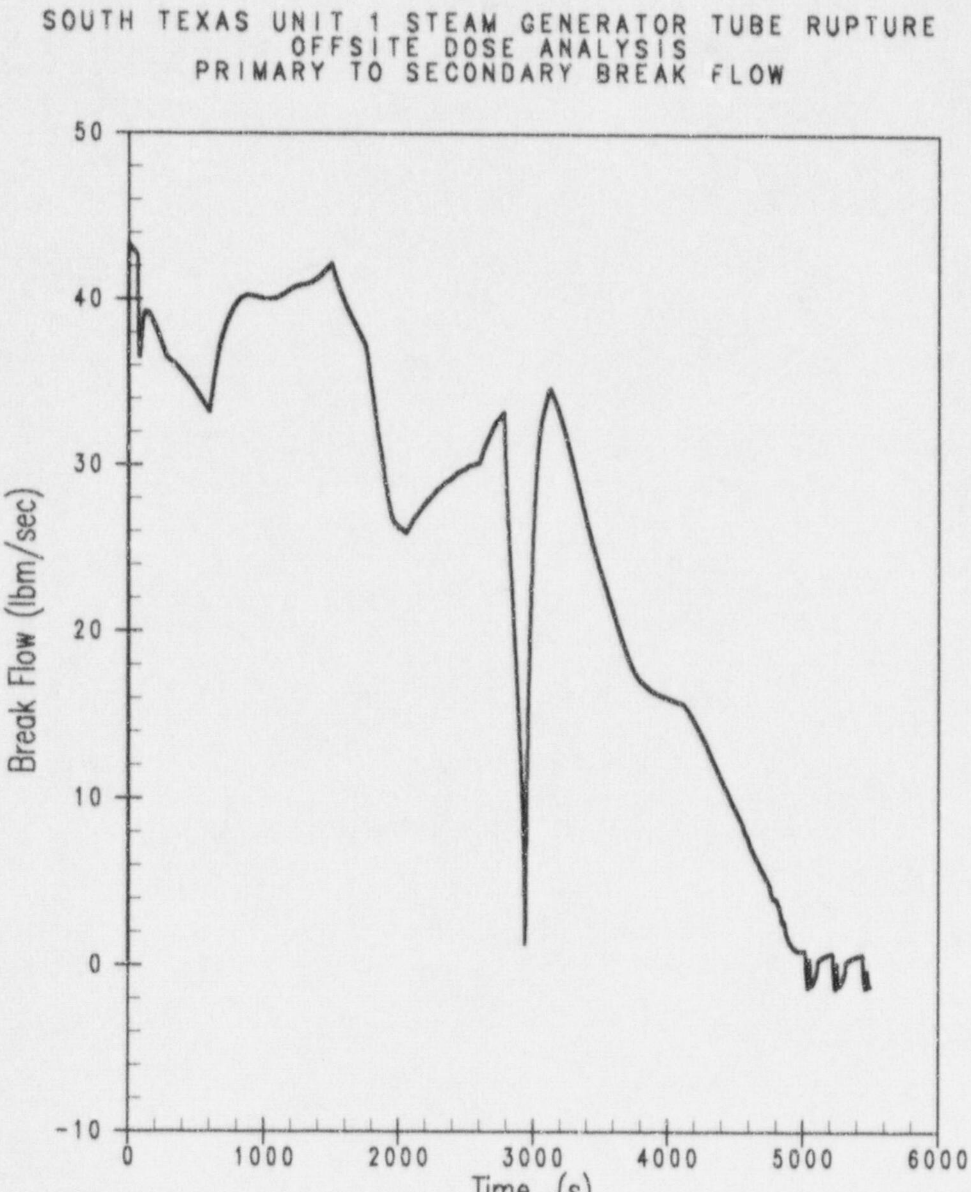
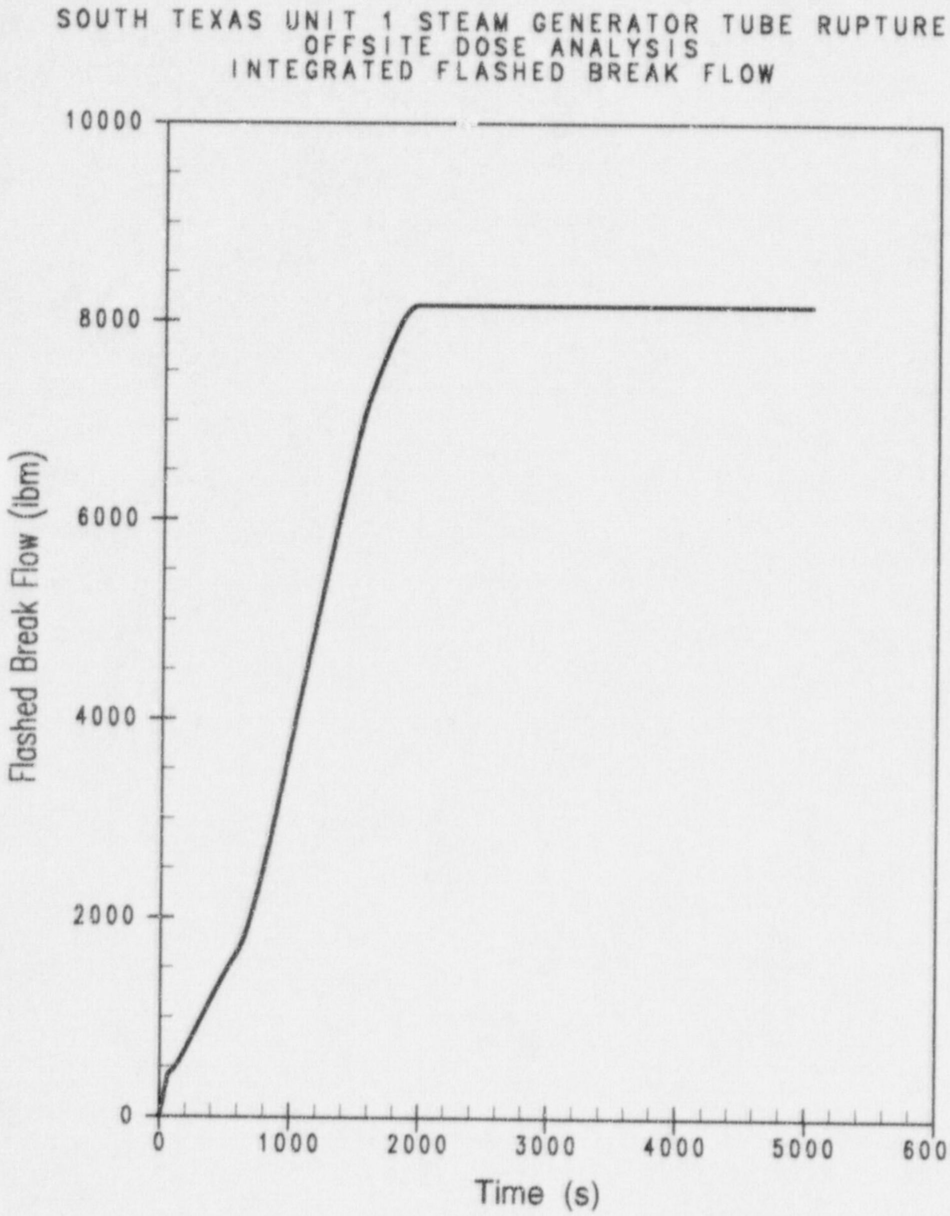

SOUTH TEXAS UNIT 1 STEAM GENERATOR TUBE RUPTURE
OFFSITE DOSE ANALYSIS
PRIMARY TO SECONDARY BREAK FLOW
Break Flow (lbm/sec)
50
40
30
20
10
0
-10
0
1000
2000
3000
4000
5000
6000
Time (s)

Figure 5.5-2


SOUTH TEXAS UNIT 1 STEAM GENERATOR TUBE RUPTURE
OFFSITE DOSE ANALYSIS
INTEGRATED FLASHED BREAK FLOW
10000
8000
6000
4000
2000
0
Flashed Break Flow (lbm)
0
1000
2000
3000
4000
5000
6000
Time (s)

Since the LOCA is the limiting accident for the control room and technical support center doses, control room and technical support center doses were not calculated for this accident.

The radiological consequences analysis makes the following assumptions:

1. Offsite power is assumed lost upon reactor trip; therefore, the condenser is not available for steam dump operation.

2. No turbine runback is assumed. This will lead to a reactor trip at a higher reactor power level. The higher power level will result in greater initial steam releases through the SG PORVs and/or safety valves increasing the consequences of the accident.

3. Prior to the accident, an equilibrium concentration of fission products exists in the primary system.

4. Prior to the accident, the secondary coolant specific activity is equal to the Technical Specification limit of 0.10 µci/gm dose equivalent I-131.

5. A primary-to-secondary leakage of 1 gal/min is assumed to continue for 36 hours following the accident at the pre-accident rates. For the Model E steam generators, it is assumed that 0.3 gal/min leakage occurs in the ruptured steam generator and the remaining 0.7 gal/min is split equally between the three intact steam generators. For the Δ94 steam generators, all of the 1 gpm leakage is assumed to occur in the three intact steam generators and no primary-to-secondary leakage (other than the ruptured tube flow) is assumed in the ruptured steam generator.

6. Eight hours after the accident, the RHRS is placed in operation to cool the plant to Cold Shutdown. The only steam release after eight hours is through the MSIV above seat drain line flow restriction orifices.

7. The iodine partition factor in the steam generators during the accident is equal to 0.01.

8. For a pre-existing iodine spike, the primary coolant concentration is assumed to be equal to the Technical Specification limit of 60 µci/gm. The secondary coolant specific activity is equal to the Technical Specification limit of 0.1 ci/g dose equivalent I-131.

9. For an accident-initiated iodine spike, the primary coolant iodine concentration is assumed to a function of time. Increasing the source term or release rate from the fuel by a factor of 500 accounts for the spike. The amount of iodine buildup in the RCS is then limited to 60µci/gm. Prior to the accident, the secondary coolant concentration is based upon equilibrium reactor coolant concentration with 1 gal/min primary-to-secondary leakage.

6. CONCLUSIONS

Neither the proposed feedwater temperature reduction for the current Model E steam generators nor the proposed replacement of the current Model E steam generators with Δ94 steam generators will result in a significant increase in doses to the public due to the accidents postulated in Chapter 15 of the Updated Final Safety Analysis Report.

7. IMPLEMENTATION

This proposed operating license amendment request should be implemented following the next Unit 1 refueling outage (1RE08). Currently, that outage is scheduled to be completed by April 29, 1999. Therefore, to allow for timely implementation of this proposed license amendment, the NRC is requested to review and approve this amendment request by June 1, 1999. Also, the South Texas Project requests that the effective date of this proposed license amendment be 30 days after the date of NRC approval.

8. REFERENCES

1. Letter from Mr. T. H. Cloninger, South Texas Project, to the U.S. Nuclear Regulatory Commission, "Unit 1 Technical Specifications 3.4.5 and 3.4.6.2" dated January 22, 1996.

2. Letter from Mr. W. T. Cottle, South Texas Project, to the U.S. Nuclear Regulatory Commission, "Revised Proposed Amendment to Incorporate Voltage-Based Repair Criteria In Unit 1 Technical Specifications 3.4.5 and 3.4.6.2", dated April 4, 1996.

3. Letter from Mr. S. E. Thomas, South Texas Project to the U.S. Nuclear Regulatory Commission, "Response to Questions Asked by EMCB and PERB Staff on Steam Generator Voltage-Based Repair Criteria Submittal (TAC M 94535)", dated May 2, 1996.

4. Letter from Mr. T. W. Alexion, U.S. Nuclear Regulatory Commission, to Mr. W. T. Cottle, South Texas Project, "South Texas Project, Unit 1 - Amendment No. 83 To Facility Operating License No. NPF-76 (TAC No. M94535)", dated May 22, 1996 (Safety Evaluation for the Voltage-Based Repair Criteria In Unit 1).

5. Report of Committee II on Permissible Dose for Internal Radiation, International Commission on Radiation Protection (ICRP) Publication 2 (1959).

6. Murphy, K.G. and Campe, K.M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19", Paper presented at the 13th AEC Air Cleaning Conference.

7. "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", USNRC Regulatory Guide 1.109, Rev. 1, October 1977.

8. International Commission on Radiation Protection (ICRP), "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, *Annals of the ICRP Volume 2*, 1979.

9. US Environmental Protection Agency, "*Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*," Federal Guidance Report No. 11, EPA-520/1-88-020, September, 1988.

10. US Environmental Protection Agency, "*External Exposure to Radionuclides in Air, Water, and Soil*," Federal Guidance Report No. 12, EPA 402-R-93-081, September 1993.

11. Department of Energy, "*External Dose-Rate Conversion Factors for Calculation of Dose to the Public*," DOE/EH-0070, July, 1988.

12. Letter from Mr. T. H. Cloninger, South Texas Project, to the U.S. Nuclear Regulatory Commission, "Proposed Amendment to Incorpoate Voltage-Based Repair Criteria into Technical Specifications 3.4.5" dated February 16, 1998.

13. Letter from Mr. T. H. Cloninger, South Texas Project, to the U.S. Nuclear Regulatory Commission, "Proposed Amendment to Incorpoate Voltage-Based Repair Criteria into Technical Specifications 3.4.5" dated April 2, 1998.

14. Letter from Mr. D. A. Leazar, South Texas Project, to the U.S. Nuclear Regulatory Commission, "Response to NRC's Request for Additional Information (TAC Nos. MA0967 and MA0968)", dated July 15, 1998.

ATTACHMENT 3

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The South Texas Project has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Part 50 Section 92 Paragraph c (10 CFR 50.92(c)), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

INTRODUCTION

The South Texas Project is pursuing two proposed modifications to the facility:

1. Reduction of the nominal feedwater temperature of the existing Model E steam generators from 440°F to 420°F

   The purpose of this change is to allow feedwater inlet temperature to the steam generators (SGs) to be operated in a range between 440°F and 420°F. The feedwater temperature reduction will be accomplished by partially opening the high pressure feedwater heater bypass valve(s). The feedwater temperature reduction will allow 100% reactor power to be achieved with degraded steam generators.

2. Replacement of the current Model E steam generators with Δ94 steam generators.

   This modification is necessary due to the condition of the current Model E steam generators in Unit 1. The proposed steam generator replacement effort is being evaluated with a combination of this submittal, internal 10 CFR 50.59 reviews, as appropriate, and other submittals for necessary changes to the facilities' Technical Specifications and operating license. The radiological analyses performed assume the feedwater temperature of the Δ94 steam generators may be as low as 390°F.

This evaluation only addresses the radiological aspects of these proposed changes. The radiological impacts to the facility are minimal. Also, due to timing considerations of the replacement steam generator project in Unit 1, and the desire to have the ability to reduce the feedwater temperature in either unit, bounding accident analyses were performed for offsite dose consequences. This submittal proposes necessary changes to the UFSAR to reflect the radiological impact of the proposed modifications.

NO SIGNIFICANT HAZARDS EVALUATION

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   This document updates the facilities' radiological design basis, as described in the Updated Final Safety Analysis Report, to address both a reduction in allowed nominal feedwater temperature for Model E steam generators from 440° F to 420° F and the replacement of Model E steam generators with Δ94 steam generators. Therefore, these changes do not change the probability of an accident previously evaluated.

   A safety analysis has been performed, including evaluations of existing analyses and performance of bounding and/or confirming calculations, to determine the impact of the proposed changes. Effects on the dose analyses due to the accompanying physical changes to the plant are slight. However, some improvements were made to the analytical models used in the analyses. These improvements were responsible for the majority of the increase in offsite doses. While the radiological consequences of some postulated accidents increased, all results remain within the acceptance criteria, as delineated in 10 CFR 100 and the Standard Review Plan (NUREG-0800).

   The radiological consequences of the postulated accidents remain within their respective acceptance criteria with the use of the revised analysis methodologies. Therefore, the change to allow operation of the Model E steam generators at a reduced feedwater temperature of 420° F and the replacement of Model E steam generators with Δ94 steam generators do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   This document updates the facilities' radiological design basis, as described in the Updated Final Safety Analysis Report, to address both a reduction in allowed nominal feedwater temperature for Model E steam generators from 440° F to 420° F and the replacement of Model E steam generators with Δ94 steam generators. Since the proposed changes to the

Updated Final Safety Analysis Report are analytical in nature, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

A safety analysis has been performed, including evaluations of existing analyses and performance of bounding and/or confirming calculations, to determine the impact of the proposed changes. Effects on the dose analyses due to the accompanying physical changes to the plant are slight. However, some improvements were made to the analytical models used in the analyses. These improvements were responsible for the majority of the increase in offsite doses. While the radiological consequences of some postulated accidents increased, all results remain within the acceptance criteria, as delineated in 10 CFR 100 and the Standard Review Plan (NUREG-0800), for the respective accidents.

The radiological consequences of the postulated accidents remain within their respective acceptance criteria with the use of the revised analysis methodologies. Therefore, the change to allow operation of the Model E steam generators at a reduced feedwater temperature of 420° F and the replacement of Model E steam generators with Δ94 steam generators do not involve a significant reduction in a margin of safety.

Based on the above evaluation, South Texas Project concludes that the proposed changes to the UFSAR involve no significant hazards consideration.

ATTACHMENT 4

UPDATED FINAL SAFETY ANALYSIS REPORT MARK-UPS

The below listed UFSAR pages are provided in this attachment in support of this amendment. Proposed revisions are indicated as appropriate.

Pages:	Pages
TC 11-1 (Table of Contents)	15.4-33*
	15.4-34*
11.1-5	15.4-35*
11.2-1	15.4-36
11.3-1	15.4-43
	15.4-44
TC 15-2 (Table of Contents)	15.4-45
TC 15-7 (Table of Contents)	
TC 15-8 (Table of Contents)	15.6-8*
TC 15-9 (List of Tables)	15.6-9*
	15.6-10
15.1-16*	15.6-13
15.1-17	
15.1-18	15.7-1*
15.1-19	15.7-2
15.1-23	15.7-3
15.1-24*	
15.1-25	15.A-2
15.3-8*	15.B-1
15.3-9	15.B-9
15.3-16	15.B-10
15.3-17	15.B-13

* Pages with no changes shown are provided to support review of the proposed License Amendment.

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11.1.4	Activity in Radwaste Systems	11.1-5
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11.1.4 Activity in Radwaste Systems

The design basis source terms for shielding and component failures for the Radwaste Systems are based upon the concentrations shown in Table 11.1-2. The expected activities of the Radwaste Systems for effluent analysis are based upon the concentrations shown in Table 11.1-7. The Liquid Radwaste System is further described in Section 11.2, the Gaseous Radwaste System in Section 11.3, and the Solid Radwaste System in Section 11.4. The shielding of these systems is described in Section 12.3.

11.1.5 Leakage Sources

The systems containing radioactive liquids are potential sources of leakage to the plant buildings and then to the environment. Leakage from the primary system to the Containment is expected to be less than 240 lb/day. This leakage comes from such sources as valve packings. Leakage from the systems located in the Mechanical Auxiliary Building (MAB) is expected to be less than 160 lb/day. This leakage comes from such potential sources as pump gland seals and valve packings. Total steam leakage in the TGB is expected to be less than 1,700 lb/hour, as discussed in Section 11.3.2.

These leakage sources and the resulting airborne concentrations are discussed more fully in Section 12.2.2.

Potential release points of radioactive effluents are discussed in Sections 11.2 and 11.3.

11.1.6 The Impact of Extended Burnup Fuel on Source Terms

The source terms presented in Sections 11.1.1 through 11.1.5 are based on an equilibrium fuel cycle using discharge burnup of 33,000 MWD/MTU. The use of extended burnup fuel at STPEGS has been reviewed in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors" (References 11.1-4, 11.1-5) and has been determined to not significantly change the results previously presented in safety analysis reports based on operation to 33,000 MWD/MTU discharge burnup.

For VANTAGE 5H fuel, source terms based on an equilibrium fuel cycle using batch average burnups of 20,000 MWD/MTU, 40,000 MWD/MTU, and 60,000 MWD/MTU (each at 1/3 core size) with fuel enriched to a nominal 5.0 w/o U-235 have been evaluated. The results do no significantly change the results in Sections 11.1.1 through 11.1.5.

INSERT 11.1-1

CN 2272

INSERT for Sections 11.1.7 and 11.1.8

Insert 11.1-1

11.1.7 The Impact of Operating at a Reduced Feedwater Temperature on Source Terms

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological source terms described in Section 11.1 has been evaluated. It was determined that operation under either scenario would have a negligible impact on the isotopic inventories presented in Section 11.1.

11.1.8 The Impact of Westinghouse Model Delta 94 Replacement Steam Generators on Source Terms

The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological source terms described in Section 11.1 has been evaluated. It was determined that operation with either type of steam generator would have a negligible impact on the isotopic inventories presented in Section 11.1.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

This section describes the design and operating features of the Units 1 and 2 Liquid Waste Processing Systems (LWPS). Total plant liquid releases from all sources are estimated and summarized in Section 11.2.3.5. The design meets the intent of Branch Technical Position (BTP) ETSB 11-1, Rev. 1.

11.2.1 Design Bases

The function of the LWPS is to collect and process radioactive liquid wastes generated from plant operation and maintenance and to reduce radioactivity and chemical concentrations to levels acceptable for discharge or recycle.

The principal design objectives of the LWPS are:

1. Collection of liquid wastes generated during anticipated plant operations which potentially contain radioactive nuclides.

2. Provision of sufficient processing capability such that liquid waste may be discharged to the environment at concentrations below the regulatory limits of 10CFR20 and consistent with the as low as is reasonably achievable (ALARA) guidelines set forth in 10CFR50, Appendix I.

X INSERT 11.2-1

CN 2272

Design considerations for shielding and the reduction of radiation exposure to personnel are given in Section 12.1.3.

Source terms used to determine shielding requirements are given in Table 11.1-2 (1 percent fuel cladding defects); dose design objectives are based upon the source terms in Table 11.1-7 (realistic basis).

Plant operational releases from the LWPS will be below regulatory and/or licensing requirements.

During operation with excessive reactor coolant leakage or temporary malfunction in the LWPS, additional and/or alternate processing capacity in the LWPS is available to limit releases to approximately the same as during normal operation.

Section 11.2.3.5 establishes that the LWPS adequately meets the above-listed design objective.

11.2.2 Systems Descriptions

11.2.2.1 General Process Descriptions. The LWPS collects and processes potentially radioactive wastes for release to the environment. Provisions are made to sample and analyze fluids before they are discharged. Based upon the laboratory analyses, these wastes are either released under controlled conditions into the discharge of the Circulating Water System (CWS) via the Open Loop Auxiliary Cooling Water System (OLACW) or retained for further processing. A permanent record of liquid releases is provided by laboratory analyses of known volumes of waste.

INSERT for Section 11.2.1

Insert 11.2-1

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation under either scenario would have a negligible impact on the isotopic inventory of the liquid waste processing system and the radiological consequences of a LWPS failure, as described in Chapter 15.7.

The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation with either type of steam generator would have a negligible impact on the isotopic inventory of the liquid waste processing system and the radiological consequences of an LWPS failure, as described in Chapter 15.7.

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

11.3.1 Design Bases

The design objectives of the Gaseous Waste Management System (GWMS) are twofold. The first objective is to process and control the release of gaseous radioactive effluents to the site environs in order to meet the requirements of 10CFR20 and the dose design objectives specified in 10CFR50, Appendix I. The second objective is to remove fission product gases from the reactor coolant and process these gases before they are released. These objectives are achieved when the input sources are as specified in Table 11.1-2 (design basis source terms). The GWMS is designed so that radiation exposure to personnel will be as low as is reasonably achievable (ALARA).

The effect of the V5H fuel upgrade on the radioactivity concentrations in the fluid systems was reviewed and it was determined that the original reactor coolant activity listed in Table 11.1-2 is bounding. Therefore, the FSAR analyses based on this activity are not adversely impacted by the fuel upgrade. The corresponding reactor core activity for the V5H upgrade is shown in Table 15.A-1A. X INSERT 11.3-1 CN 2272

Various gas treatment systems are employed for the control of noble gases and iodine. The process vents and the building ventilation air filtration systems are described in detail in Section 9.4 and are only briefly described in this section for completeness. The Gaseous Waste Processing System (GWPS) removes and processes fission product gases from the Reactor Coolant System (RCS) and other miscellaneous sources of fission product gases. Table 11.3-1.1 lists the expected activity releases. During refueling the Reactor Coolant Vacuum Degassing System (RCVDS) removes fission product gases from the RCS free space. The RCVDS is used to reduce the time between draindown and reactor head removal.

The design bases for the GWMS are as follows:

1. The GWMS is designed to limit routine station activity releases to a small fraction of the limits specified in 10CFR20, and to minimize doses to ALARA in accordance with 10CFR50, Appendix I.

2. The GWMS furnishes protection against inadvertent release of significant quantities of gaseous and particulate radioactive material to the environs by providing:

   a. Design redundancy when required.

   b. Instrumentation for detection and alarm of abnormal conditions.

   c. Procedural controls and/or provisions for automatically halting the discharge of gaseous waste effluents if their activity exceeds preset limits.

   d. Continuous monitoring of the various holdup and process systems. Integral control and monitoring instruments in the process lines preclude uncontrolled release of radioactive material to the environment.

   e. Adequate time for operator decision and action when the monitors indicate the development of abnormal conditions.

INSERT for Section 11.3.1

Insert 11.3-1

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation under either scenario would have a negligible impact on the isotopic inventory of the gaseous waste processing system and the radiological consequences of a GWPS failure, as described in Chapter 15.7.1.

The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation with either type of steam generator would have a negligible impact on the isotopic inventory of the gaseous waste processing system and the radiological consequences of a GWPS failure, as described in Chapter 15.7.1.

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Core Power and Reactor Coolant System Transient

Figures 15.1-15 through 15.1-17 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (Case A) Offsite power is assumed available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one SG. Should the core be critical at near zero power when the rupture occurs, the initiation of SI by low steam line pressure will trip the reactor. Steam release from more than one SG will be prevented by automatic closure of the fast-acting isolation valves in the steam lines via the low steam line pressure signal. Even with the failure of one valve, release is limited to no more than 10 seconds from the other SGs while the one generator blows down. The steam line stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 15.1-17 the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) shortly before boron solution at 2,800 ppm enters the RCS. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the SIS. The variation of mass flow rate in the RCS due to water density changes is included in the calculation, as is the variation of flow rate in the SIS due to changes in the RCS pressure. The SIS flow calculation includes the line losses in the system, as well as the pump head curve.

Figures 15.1-18 through 15.1-20 show the response of the salient parameters for case b which corresponds to the case discussed above with additional LOOP at the time the SI signal is generated. The SI delay time includes 10 seconds to start the SBDG and in 12 seconds the pump is assumed to be at full speed. Criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying SG to extract heat from the RCS is reduced by the decreased flow in the RCS.

It should be noted that, following a steam line break only, one SG blows down completely. Thus, the remaining SGs are still available for dissipation of decay heat after the initial transient is over. In the case of LOOP, this heat is removed to the atmosphere via the steam line safety valves.

MSLB Analysis for Above MSIV Seat Drain Line Flow Restriction Orifices

A MSLB analysis was performed to determine the effect of replacing the Above Seat Main Steam Line SOVs with 3/8" orifices. The design change has negligible affect on the RCS response. The design change also has negligible affect on the additional mass/energy release to the RCB and IVC. For the duration from MSLB initiation to cold shutdown, the plant is assumed to be at hot standby for 36 hours with instantaneous cold shutdown at 36 hours. This is conservative in evaluating the additional mass release through the orifice since cooldown occurs quickly after a MSLB, and hot shutdown can be achieved within three hours. After 36 hours, no further steam releases are assumed.

Margin to Critical Heat Flux

A DNB analysis was performed for both of these cases. It was found that the DNB design basis as stated in Section 4.4 was met for all cases.

15.1.5.3 Radiological Consequences. The postulated accidents involving release of steam from the secondary system do no result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the SGs. A conservative analysis of the potential offsite doses resulting from a steamline break outside Containment upstream of the MSIV is presented using the Technical Specification limit secondary coolant concentrations. Parameters used in the analysis are listed in Table 15.1-2.

The conservative assumptions and parameters used to calculate the activity released and offsite doses for a steam line break, assuming no iodine spike, are the following:

1. Prior to the accident, an equilibrium specific activity of radionuclides exists in the primary system. Reactor coolant concentrations remain constant following the accident (with the exception of activity added due to fuel damage). The equilibrium reactor coolant concentrations are presented in Table 15.A-2.

2. Prior to the accident, the secondary coolant specific activity is based upon equilibrium reactor coolant concentration with 1 gal/min primary-to-secondary leakage. This concentration is presented in Table 15.A-2.

~~3. The fuel rod cladding is breeched in a number of fuel rods, which results in the release to the reactor coolant of five percent of the total core gap inventory. This activity is assumed uniformly mixed in the primary coolant.~~ | CN 2272

3, ~~4.~~ The primary-to-secondary leakage of 1 gal/min (Technical Specification limit) is assumed to continue for 36 hrs following the accident. It is assumed that 0.35 gal/min leakage occurs in the defective SG and 0.217 gal/min each of the unaffected SGs. | CN 2272

4, ~~5.~~ Offsite power is lost; MS condensers are not available for steam dump. | CN 2272

5, ~~6.~~ Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. The only steam release after eight hours is through the above MSIV seat drain line flow restriction orifices. | CN 2272

6, ~~7.~~ The iodine partition factor in the SGs is the ratio of the amount of iodine per unit mass of steam to the amount of iodine per unit mass of liquid and is equal to 0.01. | CN 2272

The steam releases and meteorological parameters are given in Table 15.1-2.

If the postulated accident is assumed to occur coincident with an existing iodine spike (caused by a previous power transient), the assumptions and parameters used to evaluate the activity releases and offsite doses are unchanged, with two exceptions. The primary coolant concentrations are assumed to be equal to the Technical Specification limit for full power operation following an iodine spike. These concentrations are presented in Table 15.A-4. The secondary coolant specific activity is equal to the Technical Specification limit of 0.1 μCi/g dose equivalent I-131. This dose equivalent activity is presented in Table 15.A-5. Fuel failures due to the accident are not assumed to occur coincident with an iodine spike.

If the postulated accident is assumed to result in an iodine spike (caused by the power transient of reactor trip), the assumptions and parameters used to evaluate the activity releases and the offsite doses are again unchanged, with two exceptions. The primary coolant iodine concentrations are assumed to be functions of time. The spike is accounted for by increasing the source term or release rate from the fuel by a factor of 500. Further discussion of this iodine spiking is contained in Appendix 15.A.3. Fuel failures are not assumed to occur during the accident. Prior to the accident, the secondary coolant concentration is based upon equilibrium reactor coolant concentration with 1 gal/min primary-to-secondary leakage. This concentration is presented in Table 15.A.2.

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The thyroid, gamma and beta doses for the steamline break for the various cases analyzed are given in Table 15.1-3 for the Exclusion Zone Boundary (EZB) of 1,430 meters and the Low Population Zone (LPZ) of 4,800 meters.

15.1.5.4 Conclusions. The analysis has shown that the criteria stated in Section 15.1.5.1 are satisfied. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that no DNB occurs for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position. The radiological consequences of this event are within the guidelines of 10CFR100.

During plant start-up, the above MSIV seat drain line valves are opened for removal of accumulated condensate to protect the turbine from water induction damage and to prevent water hammer in the steam lines. During normal operations, manual valves isolate the above MSIV seat drain lines. Specific analyses for simultaneous steam releases from all four steam generators via opened above MSIV seat drain lines concurrent with a steam generator tube rupture (SGTR) event or a Main Steam Line Break with a design primary to secondary system leak demonstrates that radiological doses will not exceed 10CFR100 limits and the additional steam demand will not result in exceeding applicable reactor safety acceptance criteria. Due to the use of restricting orifices, flow from the lines will be limited and no operator action is required to close the above MSIV seat drain line isolation valves.

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Since the fuel does not experience DNB, a scenario considering fuel clad damage is not considered.

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological consequences of a main steam line break has been analyzed. The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological consequences was also analyzed.

The analysis assumes the primary-to-secondary leakage in the faulted steam generator instantly flashes to steam and is released to the environment. Also, as per the Standard Review Plan, the initial iodine concentration in the secondary side is assumed to be at the Technical Specification limit of 0.1μci/gm dose equivalent Iodine-131. Tables 15.1-2 and 15.1-3 reflect a bounding analysis for all three steam generator configurations (e.g., the Model E steam generators at a nominal feedwater temperature of 440°F, the Model E steam generators at a reduced feedwater temperature of 420 [illegible] and the Model Delta 94 steam generators at a feedwater temperature of 390°F).

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15.1.6 Voltage-Based Steam Generator Tube Repair Radiological Analysis
An analysis was performed to determine the maximum primary-to-secondary leak rate, with a concurrent main steam line break, which would result in offsite doses remaining within the limits of 10 CFR 100 and control room doses within GDC 19 limits. This analysis was performed for the implementation of voltage-based steam generator tube repair and is valid for either unit (see References 15.1-5, -6, -7, and -8). The analysis is bounding for the Model E steam generators operating at a feedwater temperature between 420° F and 440° F and for the Westinghouse Delta 94 steam generators operating at a feedwater temperature above 390° F.

For voltage-based steam generator tube repair, the Technical Specifications have been revised to lower the allowed RCS leakage from a limit of 1 gpm total leakage and 500 gallons per day from any one steam generator to a limit of 600 gpd (0.42 gpm) total leakage and 150 gallons per day from any one steam generator. The leakage limits are more restrictive than the standard operating leakage limits and are intended to provide an additional margin to accommodate a crack in a steam generator tube which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program (as described in the Technical Specifications), provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner. The lower allowable steam generator leakage limit minimizes the radiological release during a Main Steam Line Break.

If the voltage-based steam generator tube repair method is used, the end-of-cycle voltage distribution will be established for an upcoming cycle based upon the previous end-of-cycle eddy current data. Based upon this distribution, postulated steam generator tube leakage during a steam line break will be estimated. Projected leakage must remain below a level which results in offsite doses remaining within the limits of 10 CFR 100 and control room doses within GDC 19 limits. Should the leakage estimation exceed the maximum allowed primary-to-secondary leak rate, as determined by this analysis, actions will be taken to ensure the leakage estimation is reduced to an acceptable value.

Insert 15.1-2 (Continued)

A description of the MSLB analysis may be found in Section 15.1.5.3. A description of the control room HVAC may be found in Section 6.4.4.1. The following assumptions were made to determine the limiting maximum primary-to-secondary post-MSLB leak rate in the steam generator in the faulted loop:

1. The source term is based upon a power level of 4100 MW thermal, 5 w/o enrichment, and a 3 region core with equilibrium cycle core at end of life. The three regions have operated at a specific power of 39.3 MW/MTU for 509, 1018, and 1527 EFPD, respectively.

2. The accident results in no failure of fuel rod cladding.

3. Reactor coolant density is 8.33 lbs/gal (room temperature conditions).

4. The equilibrium secondary activity before the Steam Generator rupture is based upon a preexisting primary to secondary leakage of 1 gpm. This is conservative since the Technical Specifications for the voltage-based repair criteria limits the preexisting leakage to 150 gpd per steam generator or 600 gpd (0.42 gpm) total.

5. The total steam generator tube leak rate prior to the accident and until 8 hours after the start of the accident is 0.42 gpm (approx. 600 gpd). This is conservatively divided into 0.147 gpm (35%) to the affected loop and 0.273 gpm (65%) to the unaffected loops.

6. For a pre-existing iodine spike, the activity in the reactor coolant is based upon an iodine spike which has raised the reactor coolant concentration to 60 micro Ci/gm of dose equivalent I-131. The secondary coolant activity is based on 0.1 micro Ci/gm of dose equivalent I-131. Noble gas activity is based on 1% failed fuel. This scenario is less limiting than the accident-initiated iodine spike case.

7. For a accident-induced iodine spike, the accident initiates an iodine spike in the RCS which increases the iodine release rate from the fuel to a value 500 times greater than the release rate corresponding to a RCS concentration of 1 micro Ci/gm dose equivalent I-131. The iodine activity released from the fuel to the RCS is conservatively assumed to mix instantaneously and uniformly in the RCS. Noble gas activity is based on 1% failed fuel. Steam generator activities are assumed to be identical to the pre-existing iodine spike (iodides based on 0.1 micro Ci/gm of dose equivalent I-131; noble gas activity based on 1% failed fuel).

8. Following the accident, auxiliary feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. The primary-to-secondary leakage in the faulted steam generator is assumed to flash to steam and be immediately released to the atmosphere with an iodine partition factor of 1.0.

INSERT 15.1-2 (Continued)

9. Following the accident, the primary to secondary leakage in the affected steam generator is assumed to be instantaneously released to the environment. This is conservative since the free volume above the top of tubes and the volume of the 30" pipe inside the containment provide space for accumulation and hold-up of radionuclides before release to the environment. Another conservatism is that the plate-out effect was not modeled.

10. In the intact steam generators, the primary to secondary leakage is mixed into the existing mass in the steam generators. The release to the environment is modeled as a release of steam generator mass with an iodine partition factor of 0.01.

11. Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. This ceases the primary-to-secondary leakage. The only steam release after eight hours is the release of secondary side inventory through the MSIV above seat drain line flow restriction orifices. All releases stop 36 hours after the accident.

12. Offsite Power is lost; the condensers are unavailable for steam dump.

13. All activity is released to the environment with no consideration given to radioactive decay or to cloud depletion by ground deposition during transport to the exclusion zone boundary and low population zone.

14. The X/Q for the RCB to CR/TSC intake is assumed to apply for MSLB site to the CR/TSC intake.

15. The offsite, CR and TSC doses change linearly as a function of the primary to secondary break flow.

Two analyses were performed. One at zero additional leakage and the second assuming an additional 10 gpm primary-to-secondary leakage. The limiting maximum primary-to-secondary post-MSLB leak rate in the steam generator in the faulted loop was determined by a linear extrapolation on the thyroid doses. A point at approximately 90% of either the 10 CFR 100 limit (as expanded upon by the Standard Review Plan limit) or the GDC 19 limit was chosen on this "flow versus dose" curve to determine the limiting maximum primary-to-secondary post-MSLB leak rate.

A design description of the control room HVAC system and how it functions to meet its design basis may be found in UFSAR Section 6.4. Table 6.4-2 presents the HVAC parameters for the doses to control room operators due to a postulated LOCA. The voltage-based repair criteria/MSLB analysis used the data from this table. The filter efficiencies for the makeup air filters were modified to reflect an assumed single failure of a filter heater, in addition to an assumed single failure of a Standby Diesel Generator and its associated train. Also, the efficiencies were calculated assuming a flow of 2200 cfm. A design description of the technical support center HVAC system and how it functions to meet its design basis may be found in UFSAR Section 9.4.1.

The results of the analyses are given in Table 15.1-5.

REFERENCES

Section 15.1:

15.1-1 Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary Class 2), WCAP-7907-A (Proprietary Class 3), April 1984.

15.1-2 "Westinghouse Anticipated Transients Without Trip Analysis", WCAP-8330, August 1974.

15.1-3 Moody, F. S., "Transactions of the ASME, Journal of Heat Transfer", Figure 3, page 134, February 1965.

15.1-4 Hollingsworth, S. D. and Wood, D. C., "Reactor Core Response to Excessive Secondary Steam Releases", WCAP-9226, January 1978.

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15.1-5. Letter from Mr. T. H. Cloninger, South Texas Project, to the U.S. Nuclear Regulatory Commission, "South Texas Project , Unit 1, Docket No. STN 50-498, Unit 1 Technical Specifications 3.4.5 and 3.4.6.2" dated January 22, 1996.

15.1-6. Letter from Mr. W. T. Cottle, South Texas Project, to the U.S. Nuclear Regulatory Commission, "South Texas Project, Unit 1, Docket No. STN 50-498, Revised Proposed Amendment to Incorporate Voltage-Based Repair Criteria In Unit 1 Technical Specifications 3.4.5 and 3.4.6.2", dated April 4, 1996

15.1-7. Letter from Mr. S.E. Thomas, South Texas Project, to the U.S. Nuclear Regulatory Commission, "South Texas Project , Unit 1, Docket No. STN 50-498, Unit 1 Response to Questions Asked by EMCB and PERB Staff on Steam Generator Voltage-Based Repair Criteria Submittal (TAC M 94535) " dated May 2, 1996.

15.1-8 Letter from Mr. T. W. Alexion, U.S. Nuclear Regulatory Commission, to Mr. W. T. Cottle, South Texas Project, "South Texas Project, Unit 1 - Amendment No. 83 To Facility Operating License No. NPF-76 (TAC No. M94535)", dated May 22, 1996 (Safety Evaluation for the Voltage-Based Repair Criteria In Unit 1).

TABLE 15.1-2

PARAMETERS USED IN STEAM LINE BREAK ANALYSIS

Parameters			
Core thermal power, MWt		4,100 ~~3,800~~	CN 2272
SG tube leak rate prior to accident and initial 36 hrs following accident		1.0 gal/min*	
GWPS operating prior to accident		No	
Offsite power		Lost	
Fuel defects (prior to accident)		1.0%	
Primary coolant concentrations			
~~No iodine spike~~		~~Table 15.A-2~~	CN 2272
Preexisting iodine spike		Table 15.A-4	
Iodine spike caused by accident		Table 15.A-6	
Secondary coolant concentrations			
~~No iodine spike~~		~~Table 15.A-2~~	CN 2272
Preexisting iodine spike		Table 15.A-5	
Iodine spike caused by accident		Table 15.A-2	
Additional Failed fuel (following accident)		0% ~~5.0% of fuel rods in core~~	CN 2272
~~Activity released to reactor coolant from failed fuel and available for release~~		~~5% of total core gap inventory of noble gases and iodines~~	
Iodine partition factor in SGs during accident		0.01	
Initial steam release from defective SG, lbs	(0-30 min)	214,000 ~~210,000~~	CN 2272
Long term coolant release from defective SG, lbs	(0-8 hrs)	1,399 ~~1,390~~	
Steam release from three unaffected SGs, lbs	(0-2 hrs)	452,000 ~~484,000~~**	
	(2-8 hrs)	~~1,106,000~~ 1,080,000	
Steam release from the four above MSIV seat drain line flow restriction orifices	(0-36 hrs)	1.93 lbs/sec per orifice	

* 0.35 gal/min in affected SG and 0.217 gal/min per unaffected SG.

** Condenser is assumed not available for steam dump.

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TABLE 15.1-2 (Continued)

PARAMETERS USED IN STEAM LINE BREAK ANALYSIS

Parameters	
Meteorology	5 percentile Table 15.B-1
Dose model	Appendix 15.B

TABLE 15.1-3

DOSES RESULTING FROM STEAM LINE BREAK

	Exclusion Zone Boundary 1,430 m, 0-2 hrs	Outer Boundary Low Population Zone 4,800 m, Duration	Acceptance Criteria
No Iodine Spike			
Thyroid dose, rems	1.77×10^{1}	2.32×10^{1}	300
Whole-body gamma dose, rems	7.44×10^{-2}	5.23×10^{-2}	25
Skin beta dose, rems	2.71×10^{-2}	2.25×10^{-2}	25
Preexisting Iodine spike			
Thyroid dose, rems	9.63×10^{-1}	7.69×10^{-1}	300
Whole-body gamma dose, rems	1.81×10^{-3}	1.14×10^{-3}	25
Skin beta dose, rems	6.50×10^{-4}	5.74×10^{-4}	25
Iodine Spike Caused by Accident			
Thyroid dose, rems	1.81	2.33	30
Whole-body gamma dose, rems	5.32×10^{-3}	3.81×10^{-3}	2.5
Skin beta dose, rems	1.64×10^{-3}	1.50×10^{-3}	2.5

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Table 15.1-3

DOSES RESULTING FROM STEAM LINE BREAK

	Exclusion Zone Boundary 1,430 m, 0-2 hrs	Outer Boundary Low Population Zone 4,800 m, Duration	Acceptance Criteria
Pre-existing Iodine Spike			
Thyroid dose, rems	1.37	0.88	300
Whole Body dose, rems	2.78E-3	1.42E-3	25
Beta-skin dose, rems	9.66E-4	6.28E-4	25
Iodine Spike Caused by Accident			
Thyroid dose, rems	4.12	3.61	30
Whole Body dose, rems	1.36E-2	7.50E-3	2.5
Beta-skin dose, rems	3.91E-3	2.54E-3	2.5

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Insert 15.1-5

TABLE 15.1-4

PARAMETERS USED IN VOLTAGE-BASED STEAM GENERATOR TUBE REPAIR RADIOLOGICAL ANALYSIS

Parameters		
Core thermal power, MWt		4,100
SG tube leak rate for 0 - 8 hrs		0.42 gal/min*
Allowed primary-to-secondary leakage in faulted steam generator		15.4 gpm total
GWPS operating prior to accident		No
Offsite power		Lost
Fuel defects (prior to accident)		1.0%
Primary coolant concentrations		
Iodine spike caused by accident		Table 15.A-6
Secondary coolant concentrations		
Iodine spike caused by accident		Table 15.A-2
Iodine partition factor in intact SGs during accident		0.01
Initial steam release from defective SG, lbs	(0-30 min)	210,000
Steam release from three unaffected SGs, lbs	(0-2 hrs)	484,000**
	(2-8 hrs)	1,106,000
Total steam release from the MSIV above seat drain line flow restriction orifices. lbm	(0-36 hrs)	1,000,512
Control Room HVAC		See Section 6.4
TSC HVAC Parameters		
Filtered Intake Flow		1210 cfm
Unfiltered Flow		16.2 cfm
Exhaust Flow		1226.2 cfm
Intake and Recirculation Filtration Efficiencies (%)		
Particulate/Organic/Elemental		.990 for all
Volume		48170 ft^3
Meteorology		5 percentile Table 15.B-1
Dose model		Appendix 15.B
Dose Conversion Factors		ICRP 30, Table 15.B-3

* 0.147 gal/min in affected SG and 0.091 gal/min per unaffected SG.

** Condenser is assumed not available for steam dump.

Insert 15.1-5 (Continued)

TABLE 15.1-5

DOSES RESULTING FROM VOLTAGE-BASED STEAM GENERATOR TUBE REPAIR RADIOLOGICAL ANALYSIS (REM)

	Exclusion Zone Boundary 1,430 m, 0-2 hrs	Outer Boundary Low Population Zone 4,800 m, Duration	Acceptance Criteria
Iodine Spike Caused by Accident[1]			
Thyroid dose (CDE[2]), rems	15	27	30
Whole Body dose (DDE[3]), rems	0.0503	0.0491	2.5
Beta-skin dose (SDE[4]), rems	0.0274	0.0282	2.5

	Control Room (0-30 days)	Technical Support Center (0-30 days)	Acceptance Criteria
Iodine Spike Caused by Accident[1]			
Thyroid dose (CDE), rems	5.6	7.7	30
Whole Body dose (DDE), rems	0.00429	0.00259	5
Beta-skin dose (SDE), rems	0.134	0.138	30

[1] Allowed primary-to-secondary leakage in faulted steam generator is 15.4 gpm total.
[2] Committed Dose Equivalent per ICRP 30.
[3] Deep Dose Equivalent per ICRP 30.
[4] Shallow Dose Equivalent per ICRP 30.

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where:

w = amount reacted, mg/cm²

t = time, sec

T = temperature, °K

The reaction heat is 1510 cal/gm.

The effect of zirconium steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Locked Rotor with Four Loops Operating, Loss of Power to the Remaining Pumps

The transient results for this case are shown on Figures 15.3-17 through 15.3-20. The results of these calculations are also summarized in Table 15.3-2a. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted allowable stress limits. Also, the peak clad surface temperature is considerably less than 2,700°F. Both the peak RCS pressure and the peak clad surface temperature for this case are similar to the four-loop transient with power available as discussed above. The number of rods in DNB was conservatively calculated as less than 10 percent of the total rods in the core.

The calculated sequence of events is shown in Table 15.3-1.

15.3.3.3 Radiological Consequences. The postulated accidents involving release of steam from the secondary system do not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the SGs. A conservative analysis of the potential offsite doses resulting from a RCP shaft seizure accident is presented using the Technical

Specification limit secondary coolant concentrations. Parameters used in the analysis are listed in Table 15.3-3.

The conservative assumptions and parameters used to calculate the activity released and offsite doses for a pump shaft seizure accident are the following:

1. Prior to the accident, the primary coolant concentrations are assumed to be equal to the Technical Specification limit for full power operation following an iodine spike (I-131 equivalent of 60 μCi/g). These concentrations are presented in Table 15.A-4.

2. Prior to the accident, the secondary coolant specific activity is equal to the Technical Specification limit of 0.10 μCi/gm dose equivalent I-131. This dose equivalent specific activity is presented in Table 15.A-5.

3. Ten percent of the total core fuel cladding is damaged, which results in the release to the reactor coolant of 10 percent of the total gap inventory of the core. This activity is assumed uniformly mixed in the primary coolant. A second analysis with a release to the reactor coolant of 15 percent of the total gap inventory of the core is included to bound the release.

4. The primary-to-secondary leakage of 1 gal/min (Technical Specification limit) is assumed to continue for 8 hrs following the accident.

5. Offsite power is lost; MS condensers are not available for steam dump.

6. Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. No further steam or activity is released to the environment.

7. The iodine partition factor in the SGs is equal to 0.01. INSERT 15.3-1 CN 2272

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The steam releases and meteorological parameters are given in Table 15.3-3.

The thyroid, gamma and beta doses for the RCP shaft seizure accident are given in Table 15.3-4 for the Exclusion Zone Boundary (EZB) of 1,430 meters and the Low Population Zone (LPZ) of 4,800 meters.

15.3.3.4 Conclusions. Since the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2,700°F and the amount of zirconium water reaction is small, the core will remain in place and intact with no loss of core cooling capability.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description. The accident is postulated as an instantaneous failure of an RCP shaft, such as discussed in Section 5.4. Flow through the affected RCL is rapidly reduced,

INSERT for Section 15.3.3.3

Insert 15.3-1

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological consequences of a locked rotor accident has been analyzed. The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological consequences was also analyzed.

The analysis presented in Tables 15.3-3 and 15.3-4 bounds: (1) Model E steam generators at a nominal feedwater temperature of 440°F; (2) Model E steam generators at a reduced feedwater temperature of 420°F; and (3) Delta 94 steam generators at a feedwater temperature as low as 390°F. The analysis for the Model E steam generators at a nominal feedwater temperature of 440°F yields the bounding thyroid dose.

TABLE 15.3-3

PARAMETERS USED IN REACTOR COOLANT PUMP SHAFT SEIZURE ACCIDENT ANALYSIS

Parameters	
Core thermal power, MWt	4100 ~~3,800~~
SG tube leak rate prior to accident and initial 8 hrs following accident	1.0 gm
GWPS operating prior to accident	No
Offsite power	Lost
Fuel defects	1.0%
Primary coolant concentrations	Table 15.A-4
Secondary coolant concentrations	Table 15.A-5
Failed fuel (following accident)	10% of fuel rods in core
Acitivity released to reactor coolant from failed fuel and available for release	10% of total gap inventory of noble gases and iodines*
Iodine partition factor in SGs during accident	0.01
Steam release from four SGs, lb	614,000 ~~640,000~~** (0-2 hr) 1,264,000 ~~1,120,000~~ (2-8 hr)
Meteorology	5 percentile Table 15.B-1
Dose model	Appendix 15.B

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* A bounding analysis has also been performed assuming 15% failed fuel and gap inventory.

** Condensers assumed unavailable for steam dump.

TABLE 15.3-4

DOSES RESULTING FROM REACTOR COOLANT PUMP SHAFT SEIZURE ACCIDENT

	Exclusion Zone Boundary 1430 m, 0-2 hrs	Low Population Zone 4800 m, Duration
10% Total Gap Inventory		
Thyroid dose, rems	1.1	1.4
Whole body gamma dose, rems	3.8×10^{-2}	2.2×10^{-2}
Skin beta dose, rems	2.1×10^{-2}	1.3×10^{-2}
15% Total Gap Inventory		
Thyroid dose, rems	1.6	2.1
Whole body gamma dose, rems	5.7×10^{-2}	3.4×10^{-2}
Skin beta dose, rems	3.1×10^{-2}	1.9×10^{-2}

REPLACE TABLE WITH INSERT 15.3-2

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INSERT for Section 15.3.3.3

Insert 15.3-2

TABLE 15.3-4

DOSES RESULTING FROM REACTOR COOLANT PUMP SHAFT SEIZURE ACCIDENT

	Exclusion Zone Boundary 1,430 m, 0-2 hrs	Low Population Zone 4,800 m, Duration
10% Total Gap Inventory		
Thyroid dose, rems	1.1	1.6
Whole Body dose, rems	0.040	0.030
Beta-skin dose, rems	0.021	0.020
15% Total Gap Inventory		
Thyroid dose, rems	1.6	2.3
Whole Body dose, rems	0.057	0.050
Beta-skin dose, rems	0.031	0.030

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Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10 percent of the rods entered DNB based upon a detailed three-dimensional THINC analysis (Ref. 15.4-10). Although limited fuel melting at the hot spot was predicted for the full power cases, it is highly unlikely that melting will occur since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Ref. 15.4-10). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are undermoderated, and bowing will tend to increase the undermoderation at the hot spot. Since the 17 x 17 fuel design is also undermoderated, the same effect would be observed. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator-to-fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would, therefore, be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences. An analysis of the effects of a postulated rod ejection accident is performed using the assumptions of Regulatory Guide (RG) 1.77. The parameters used for the analysis are listed in Table 15.4-4.

For the analysis, it is assumed that prior to the postulated accident, the plant is operating at an equilibrium level of radioactivity in the primary and secondary systems as a result of coincident fuel defects and SG tube leakage. Following a postulated rod ejection accident, two activity release paths contribute to the total radiological consequences of the accident. The first

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release path is via Containment leakage resulting from release of activity from the primary coolant to the Containment. The second path is the contribution of steam in the secondary system dumped through the safety valves since offsite power is assumed to be lost.

15.4.8.3.1 Model: It is assumed that prior to the accident the plant has been operating with simultaneous fuel defects and SG tube leakage for a period of time sufficient to establish equilibrium levels of activity in the primary and secondary coolant. These concentrations are indicated in Table 15.4-4.

The model for the activity available for leakage from the Containment assumes that the activity in the fuel pellet-clad gap and the activity released due to fuel melting is instantaneously mixed in the Containment and available for release. The clad gap activity is assumed to be 10 percent of the iodines, 30 percent of the Kr-85, and 10 percent of the noble gases accumulated at the end of core life. All of the gap activity of the fuel rods failed by accident is assumed released to the Containment. Of the fuel melted, 100 percent of the noble gases and 25 percent of the iodines are assumed available for leakage from the Containment. The only removal processes considered for the Containment are radioactive decay and leakage.

The model for the activity available for release to the atmosphere from the safety valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that activity leaking from the primary coolant through the SG tubes following the accident. The primary coolant activity after the accident is assumed to be composed of the equilibrium activity prior to the accident, plus 100 percent of the noble gases and iodines released by fuel failed during the accident, plus 100 percent of the noble gases and 50 percent of the iodines released by fuel melted by the accident. The leakage of primary coolant to the secondary side of the SG is assumed to continue at its initial rate, assumed to be the same rate as the leakage prior to the accident, until the pressures in the primary and secondary systems are equalized. No mass transfer from the primary system to the secondary system is assumed thereafter. In case of coincident loss of offsite power, activity is assumed to be released to the atmosphere through the SG safety valves.

15.4.8.3.2 Assumptions for the Analysis: Conservative assumptions were used in the analysis of the release of radioactivity to the environment in the event of a postulated rod ejection accident. A summary of the parameters used in the analysis is given in Table 15.4-4. The upper limit of fission product release from the core for the analysis was determined using the following assumptions:

1. One hundred percent of the noble gases and iodines in the clad gaps of the fuel rods experiencing clad damage (assumed to be 10 percent of the rods in the core) is assumed released. The accident evaluation conservatively assumes this activity to be released twice: to the Containment for leakage to the atmosphere and to the primary coolant for leakage to the secondary system.

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2. The fraction of fuel melting was assumed to be 0.25 percent of the core as determined by the following method:

   a. A conservative upper limit of 50 percent of rods experiencing clad damage may experience centerline melting (a total of 5 percent of the core).

   b. Of the rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the volume actually melts (0.5 percent of the core could experience melting).

   c. A conservative maximum of 50 percent of the axial length of the rod would experience melting due to the power distribution (0.5 of the 0.5 percent of the core = 0.25 percent of the core).

Only centerline melting could occur and for a maximum time period of 6 seconds. Again the accident evaluation conservatively assumes this activity to be released twice. One hundred percent of the noble gases and 25 percent of the iodines are assumed released to the Containment and available for leakage to the atmosphere. One hundred percent of the noble gases and 50 percent of the iodines are assumed released to the primary coolant and available for leakage to the secondary system.

The following conservative assumptions were used in the analysis of the release of radioactivity to the environment in the event of a postulated rod ejection accident:

1. The activity in the fuel clad gaps from 10 percent of the fuel rods is assumed released as a result of clad damage from the accident.

2. One quarter percent of the core experiences fuel melting.

3. The activity released to the Containment through the rupture in the reactor vessel head is assumed to be mixed instantaneously through the Containment. This activity consists of 100 percent of the noble gases and iodines in the clad gap of the rods failed by the accident, plus 100 percent of the noble gases and 25 percent of the iodines in the fuel melted by the accident.

4. No credit is assumed for removal of iodine in the Containment due to Containment sprays.

5. The Containment leaks for the first 24 hrs at its design leak rate of 0.3 percent per day. Thereafter, the Containment leak rate is 0.15 percent per day.

6. One hundred percent of the noble gases and iodines in the gap of the fuel failed by the accident, plus 100 percent of the noble gases and 50 percent of the iodines in the fuel melted by the accident, is assumed mixed throughout the reactor coolant instantaneously.

7. Primary and secondary system pressures are equalized after 1,250 seconds, thus terminating primary-to-secondary leakage in the SGs.

8. ~~For the case of loss of offsite power, a total of 72,300 pounds of steam is discharged from the secondary system through the safety valves for 300 seconds following the accident. Steam release is terminated after this time.~~ INSERT 15.4-1 | CN 2272

9. All releases to the atmosphere are assumed to be at ground level.

These parameters are summarized in Table 15.4-4. INSERT 15.4-2 | CN 2272

15.4.8.3.3 Results: The thyroid, gamma and beta doses for the control rod ejection accident are given in Table 15.4-5 for the exclusion zone boundary (EZB) of 1,430 meters and the low population zone (LPZ) of 4,800 meters.

15.4.8.4 Conclusions: Conservative analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analysis has demonstrated that the fission product release, as a result of a number of fuel rods entering DNB, is limited to less than 10 percent of the fuel rods in the core. The radiological consequences of this event are well within the guidelines of 10CFR100.

15.4.9 Spectrum of Rod Drop Accidents in a BWR

Not applicable to STPEGS.

INSERT for Section 15.4.8.3.2

Insert 15.4-1

8. For the case of loss of offsite power, a total of 1.56×10^7 pounds of steam is discharged from the secondary system through the safety valves for 4500 seconds following the accident. Steam release is terminated after this time. The minimum time to release the initial steam generator mass is 191 seconds. The rate of release necessary to release the total steam generator mass of 659,000 pounds in 191 seconds is 207,000 lbm/min. Assuming this flow rate is constant for 4500 seconds yields a total mass release of 1.56×10^7 pounds.

Insert 15.4-2

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological consequences of a control rod ejection accident has been analyzed. The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological consequences was also analyzed. Both cases are bounded by the analysis presented in this section.

Table 15.4-4

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSIS

Parameters	
Core thermal power, MWt	4,100
Fuel defects prior to accident	1.0%
SG tube leak prior to and during accident	1.0 gal/min
Primary coolant concentrations	Table 15.A-2
Secondary coolant concentrations	Table 15.A-5 equivalent to Tech. Spec. limit)
Primary coolant mass, lbs	576,000
Assumed gap inventory	Table 15.A-1 (adjusted for 45,000 MWD/MTE discharge burnup)
Fuel failed by accident	10% of fuel rods in core
Fuel melted by accident	0.25% of core
Release to Containment (available for leakage)	
from fuel failed	100% of gap inventory of noble gases and iodines
from fuel melted	100% of noble gases and 25% of iodines
Containment free volume, ft^3	3.41 ~~3.2~~ x 10^6 | CN 2272
Containment leak rate, % per day	(Based on a containment free volume of 3.41 x 10^6 ft^3)
0-24 hrs.	0.30
1-30 days	0.15
Activity released to primary coolant	
from fuel failed	100% of gap inventory of noble gases and iodines
from fuel melted	100% of noble gases and 50% of iodines
Time between accident and equalization of primary and secondary system pressures, sec	~~1,250~~ 4500 | CN 2272

Table 15.4-4 (Continued)

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSIS

Parameters	
Offsite power	Lost
Partition factor in SGs during accident (for iodines)	0.01
Steam released through safety valves, lb*	~~72,300~~ 1.56×10^7
Meteorology	5 percentile Table 15.B-1
Dose model	Appendix 15.B
TOTAL steam generator mass, lbm	659,000
Minimum time to release initial steam generator mass, sec	191

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* The condenser is assumed to be unavailable for steam dump.

Table 15.4-5

DOSES RESULTING FROM ROD EJECTION ACCIDENT

Containment Leakage Contribution			
EZB distance* 0-2 hr doses			
Thyroid, rem	28.8	~~3.57 x 10^{1}~~	CN 2272
Whole body gamma, rem	0.1	~~1.2 x 10^{-1}~~	
Skin beta, rem	0.04	~~4.0 x 10^{-2}~~	
LPZ distance* accident duration doses			
Thyroid, rem	37.4	~~4.09 x 10^{1}~~	CN 2272
Whole body gamma, rem	0.07	~~7.6 x 10^{-2}~~	
Skin beta, rem	0.03	~~2.9 x 10^{0}~~	
Secondary System Release Contribution			
EZB distance, 0-2 hr doses			
Thyroid, rem	1.3	~~1.0~~	CN 2272
Whole body gamma, rem	0.5	~~5.09 x 10^{-1}~~	
Skin beta, rem	0.2	~~1.7 x 10^{0}~~	
LPZ distance accident duration doses			
Thyroid, rem	0.4	~~2.9 x 10^{-1}~~	CN 2272
Whole-body gamma, rem	0.2	~~1.49 x 10^{-1}~~	
Skin beta, rem	0.05	~~5.09 x 10^{-4}~~	

* Exclusion Zone Boundary distance is 1,430 meters.
Outer boundary of Low Population Zone is 4,800 meters from plant.

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With a LOOP:

1. Manually regulate AFW flow to the intact SGs to maintain a minimum on-scale water level. Identify the ruptured SG by uncontrolled rising water level or radiation monitors. Manually regulate AFW flow to the ruptured SG to maintain minimum water level with the SG tubes covered.

2. The SG pressure will rapidly increase, resulting in steam discharge to the atmosphere through the SG PORVs and/or safety valves.

3. Close the MSIV and isolate all other steam paths from the ruptured SG.

4. Dump steam through the intact SGs PORVs at the maximum rate to establish subcooling margin for RCS depressurization.

5. Decrease RCS pressure by use of the auxiliary pressurizer spray valves until the appropriate criteria based on pressurizer water level, RCS pressure, and/or RCS subcooling are met. This will decrease the pressure differential between the RCS and ruptured SG.

6. Based upon pressurizer water level, secondary heat sinks, RCS subcooling, and RCS pressure, stop SI pumps and control charging flow to minimize break flow to the secondary system. At this point, RCS pressure and ruptured SG pressure should be maintained approximately equal.

7. Continue dumping steam from the intact SGs and decrease RCS pressure. Decrease pressure in ruptured SG by backfill, blowdown, or steam release.

8. Initiate operation or the RHRS when the RCs temperature is less than 350°F and the RCS pressure is less than 350 psig.

The condensate accumulated in the secondary system can be examined and processed as required.

15.6.3.3 Analysis of Effects and Consequences.

Method of Analysis

The time required to terminate the break flow is determined by a STP specific simulation of the accident using the LOFTTR2 computer code. The simulation accounts for STP specific operator action times and equipment response characteristics according to the methodology detailed in WCAP-10698-P-A and Supplement 1 to WCAP-10698-P-A. In estimating the mass transfer from the RCS through the broken tube the following assumptions are made:

1. Reactor trip occurs automatically as a result of overtemperature delta T or low pressurizer pressure. An SI signal occurs as a result of low pressurizer pressure.

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2. After reactor trip, the break flow reaches equilibrium at the point where incoming SI flow is balanced by outgoing break flow. The break flow continues until the reactor operators cooldown and depressurize the RCS. The actions controlling the reactor operators response are contained in the STP Emergency Operating Procedures.

3. The SGs are controlled at the PORV setting, or the safety valve setting if it is reached.

Two analyses are performed for this accident. The first analysis determines the margin of the STP Steam Generators to an overfill condition. The second analysis determines the radiological consequences of the accident. Parameters used in the radiological consequences analysis are give in Table 15.6-3.

Margin-to-Overfill Analysis

The margin-to-overfill analysis makes the following assumptions:

1. Offsite power is assumed lost upon reactor trip.
2. The initial SG water mass is 152567 lbm, which reflects a conservative adjustment to provide the minimum margin to overfill.
3. A turbine runback is assumed to occur prior to reactor trip. This results in an increase in the ruptured steam generator's water level and thus decreases the margin to overfill.
4. Operation of the AFW system is assumed to occur. The maximum potential AFW flow rate of 675 gpm was used for each AFW pump.
5. The condenser is not available for steam dump operation.

A SGTR Analysis for Above MSIV Seat Drain Line Flow Restrictor Orifices

A SGTR analysis was performed to determine the effect of replacing the Above Seat Main Steam Line SOVs with 3/8" orifices. It is determined that for a SGTR, an additional mass release through the orifice from the ruptured and intact Steam Generators is observed. For the duration from SGTR initiation to cold shutdown, the plant is assumed to be at hot standby for 36 hours with instantaneous cold shutdown at 36 hours. This is conservative in evaluating the additional mass release through the orifice since with operator action and Limiting Conditions for Operations the secondary pressure would decrease so as to be in hot shutdown within at least 12 hours and cold shutdown within the next 24 hours. After 36 hours, no further steam releases are assumed.

Radiological Consequences Analysis

The steam releases and meteorological parameters for the steam generator tube rupture (SGTR) are given in Table 15.6-3.

If the postulated accident is assumed to occur coincident with an existing iodine spike, the primary coolant concentrations are assumed to be equal to the Technical Specification limit for full power operation following an iodine spike. These concentrations are presented in Table 15.A-4.

If the postulated accident is assumed to result in an iodine spike, the primary coolant iodine concentrations are modeled by increasing the release rate from the fuel by a factor of 500 over the initial primary system release rate. Further discussion of this iodine spike is contained in Appendix 15.A.3.

The dose models given in Appendix 15.B were used to calculate the doses resulting from a postulated SG tube rupture.

The radiological consequences analysis makes the following assumptions:

1. Offsite power is assumed lost upon reactor trip.
2. The initial SG water mass is 108,027 lbm, which reflects a conservative adjustment to provide a conservative estimate of offsite doses.

3. No turbine runback is assumed. This will lead to a reactor trip at a higher reactor power level. the higher power level will result in greater initial steam releases through the SG PORVs and/or safety valves increasing the consequences of the accident. The time of reactor trip is based on the response characteristics of the STP Reactor Protection System.
4. A minimum AFW flow of 500 GPM for each AFW pump is assumed.
5. The condenser is not available for steam dump operation.
6. Prior to the accident, an equilibrium concentration of fission products exists in the primary system.
7. Prior to the accident, the secondary coolant specific activity is equal to the Technical Specification limit of 0.10 μCi/gm dose equivalent of I-131. This activity is presented in Table 15.A-5.
8. A primary-to-secondary leakage of 1 gal/min (Technical Specification limit) is assumed to continue for 36 hrs following the accident at the pre-accident rates. It is assumed that 0.3 gal/min leakage occurs in the ruptured SG and the remaining 0.7 gal/min is split equally between the three intact steam generators.
9. Eight hours after the accident, the RHRS is placed in operation to cool the plant to Cold Shutdown. The only steam release after eight hours is through the above MSIV Seat drain line flow restriction orifices.
10. The iodine partition factor in the SGs during the accident is equal to 0.01.

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15.6.3.4 Results

Margin-to-Overfill Results. The margin-to-overfill analysis shows that the ruptured SG fluid volume increases to less than 7000 ft^3. This is less than the total SG volume of 7983 ft^3. Figure 15.6-4 shows the ruptured steam generator volume versus time. Therefore, overfill of the ruptured steam generator will not occur for a design basis SGTR for the South Texas Project.

Radiological Consequences Analysis. The results of the radiological consequences analysis are presented in Table 15.6.4. This table shows the thyroid, whole-body gamma, and beta-skin doses for the various cases analyzed for the exclusion zone boundary (EZB) distance of 1,430 meters and the low population zone (LPZ) distance of 4,800 meters. The results show that the doses are a small fraction of the 10 CFR 100 limits when an iodine spike is caused by the accident. The results also show that the dose are within the 10 CFR 100 limits when a preexisting iodine spike exists.

15.6.3.5 Conclusion. A SGTR will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed even assuming simultaneous LOOP. The event will not result in the overfilling of any steam generator. Radiological consequences from this event are within the limits of 10 CFR 100.

During plant start-up, the above MSIV seat drain line valves are opened for removal of accumulated condensate to protect the turbine from water induction damage and to prevent water hammer in the steam lines. During normal operations, manual valves isolate the above MSIV seat drain lines. Specific analyses for simultaneous steam releases from all four steam generators via opened above MSIV seat drain lines concurrent with a steam generator tube rupture (SGTR) event or a Main Steam Line Break with a design primary to secondary system leak demonstrates that radiological doses will not exceed 10CFR100 limits and the additional steam demand will not result in exceeding applicable reactor safety acceptance criteria. Due to the use of restricting orifices, flow from the lines will be limited and no operator action is required to close the above MSIV seat drain line isolation valves.

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

Not applicable to STPEGS.

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Insert 15.6-1

11. Operators are assumed to identify the ruptured steam generator and attempt to close the power operated relief valve (PORV) on the ruptured steam generator in 10 minutes. However, the PORV is assumed to fail open (the single failure for this accident scenario) at that time. It is assumed that the failed PORV is isolated by manually closing the PORV block valve within 15 minutes of the PORV failure. Therefore, the steam release via the ruptured steam generator's PORV is assumed to continue for a total of 25 minutes.

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological consequences of a steam generator tube rupture has been evaluated. Also, the impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological consequences was analyzed. Both cases are bounded by the analysis presented in this section.

The reflood phase of the transient is defined as the time period lasting from the end of refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and the beginning of reflood, the SI accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low-head and high-head safety injection pumps aid the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady-state levels associated with dissipation of residual heat generation. After the water level of the RWST reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by automatically switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the Containment emergency sumps by the low-head and high-head safety injection pumps and returned to the RCS cold legs. The Containment Spray System (CSS) continues to operate (drawing water from the sumps) to further reduce Containment pressure. Approximately 6.5 hours after initiation of the LOCA the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Description of Small Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small break LOCA, there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long-term recirculation.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-25.

15.6.5.3 Environmental Consequences. The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a postulated LOCA do not result in doses which exceed the limits specified in 10CFR100. INSERT 15.6-2 CN 2242

Dose contributions from three different sources are considered: Containment leakage, leakage from Engineered Safety Feature (ESF) components, and purging of the Containment prior to isolation. The parameters used for these analyses are summarized in Table 15.6-10.

15.6.5.3.1 Containment Leakage Contribution: Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS keeps cladding temperatures well below melting, and limits zirconium-water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus a fraction of the fission products accumulated in the pellet-cladding gap may be released to the RCS and thereby to the Containment.

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The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological consequences of a large break LOCA has been analyzed. The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological consequences was also analyzed. Both cases are bounded by the analysis presented in this section.

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15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

A number of events have been postulated which could result in a radioactive release from a subsystem or component. These events are:

1. Waste gas system failure

2. Postulated radioactive releases due to liquid-containing tank failure (release to atmosphere)

3. Postulated radioactive releases due to liquid-containing tank failure (ground release)

4. Design basis fuel handling accidents

5. Spent fuel cask drop accident

The above events are considered to be American Nuclear Society (ANS) Condition III events, with the exception of the fuel handling accidents, which are considered to be Condition IV events.

15.7.1 Waste Gas System Failure

15.7.1.1 Identification of Causes and Accident Description. The Gaseous Waste Processing System (GWPS) is designed to remove fission product gases from the reactor coolant and other miscellaneous sources and process these gases before they are released to the environment. The GWPS processes these gases through a guard bed, two charcoal delay tanks, and a high-efficiency particulate air filter before release, providing delay time for noble gas activity and ample charcoal for iodine removal.

The Reactor Coolant Vacuum Degassing System (RCVDS) is designed to remove fission product gases from the Reactor Coolant System (RCS) free space prior to reactor head removal for refueling operations. The RCVDS stores these gases in decay tanks, providing sufficient delay time for decay of noble gas and iodine activity before release to the environment.

Gaseous releases from the GWPS or the RCVDS may occur or be postulated to occur as a result of leaks in piping, leaks in vessels and other equipment, and failure of vessels or other equipment. The most limiting of these is the rupture of a GWPS charcoal adsorber tank, providing a large break area for release of activity.

The GWPS guard bed and the two charcoal adsorber tanks are designed to seismic Category I requirements; the remainder of the system is nonseismic. The Mechanical Auxiliary Building (MAB), in which the equipment is located, is, however, a seismic Category I structure. The GWPS system is classified as non-nuclear safety.

The design parameters and description of the GWPS are presented in Section 11.3. Equipment and tanks are designed for significantly higher temperatures and pressures than are expected during operation. Because of the conservative design of the GWPS components, an uncontrolled and unexpected rupture of a tank is considered improbable. A waste gas system failure is classified as an ANS Condition III event, an infrequent fault.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-26.

15.7.1.2 Analysis Assumptions. Both the GWPS and the RCVDS were analyzed. The GWPS was determined to be the limiting case. The accident conservatively assumes that a full degassing of the primary system has just occurred (i.e., the beds contain one RCS volume of iodine in addition to the average level) and that 1 percent of the iodines in the charcoal and 100 percent of the noble gases are released. In addition, degassing of the volume control tank is assumed to continue releasing gases through the break for 30 minutes. The parameters used for the analysis are summarized in Table 15.7-1.

The postulated tank rupture would release activity to the atmosphere of the MAB. It is conservatively assumed, for the purposes of this analysis, that the entire activity released by the GWPS due to the accident is released to the outside atmosphere and the environment over a 2-hour period. The meteorological parameters and the dose model used are given in Appendix 15.B.

15.7.1.3 Radiological Consequences. The doses calculated due to this postulated accident are presented in Table 15.7-1 for the exclusion zone boundary (EZB) of 1,430 meters and the outer boundary of the low population zone (LPZ) of 4,800 meters. All doses are well within the limits specified in 10CFR100. X INSERT 15.7-1 CN 2272

15.7.2 Postulated Radioactive Releases due to Liquid-Containing Tank Failure (Release to Atmosphere)

15.7.2.1 Identification of Causes and Accident Description. Radioactive liquid releases may occur, or be postulated to occur, as a result of leaks in piping, leaks in tanks and other equipment, and failure of tanks and other equipment. The most limiting of these is found to be rupture of the Recycle Holdup Tank (RHT), resulting in the release of the liquid contents to the floor of the cubicle.

The RHT has a non-nuclear safety design classification, and is designed as nonseismic; however, it is located in a seismic Category I structure.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-27.

15.7.2.2 Analysis Assumptions. For the purpose of evaluating the offsite consequences of the rupture of a storage tank, all tanks in the Liquid Waste Processing System (LWPS) and the RHT were considered. The tank containing the highest iodine inventory was found to be the RHT. Thus, the does consequences of a failure of one RHT is analyzed. The parameters used in the analysis are summarized in Table 15.7-2.

A tank rupture would release gaseous activity to the atmosphere in the MAB. It is conservatively assumed that the entire activity is released to the environment in a 2-hour period following the tank rupture. The iodine activity released to the atmosphere and the meteorological parameters used are presented in Table 15.7-2 for the analysis.

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The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation under either scenario would have a negligible impact on the isotopic inventory of the gaseous waste processing system, as described in Section 11.3, and on the radiological consequences of a GWPS failure.

The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation with either type of steam generator would have a negligible impact on the isotopic inventory of the gaseous waste processing system, as described in Section 11.3, and on the radiological consequences of a GWPS failure.

15.7.2.3 Radiological Consequences. The offsite doses calculated to result from the rupture of the RHT are presented in Table 15.7-2. The doses are seen to be a small fraction of the 10CFR100 values. INSERT 15.7-2

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15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failure (Ground Release)

15.7.3.1 Identification of Causes and Accident Description. Radioactive liquid releases may occur, or be postulated to occur, as a result of leaks in piping, leaks in tanks and other equipment, and failure of tanks and other equipment.

An analysis was performed to determine the worst possible tank failure based upon contained activity and volume. As a result of this review, the worst activity for the radionuclides considered was the evaporator concentrates tank (ECT) for radionuclides other than tritium and the RHT for tritium. It should be noted that both of these tanks are located in a seismic Category I structure.

15.7.3.2 Analysis Assumptions. For the purpose of this analysis, all tanks containing radioactivity were reviewed, taking into consideration their specific activity as well as the tank volume. The worst total activity available for release was found to be the evaporator concentrates tank for all nuclides except tritium. Due to its size, the RHT contains the most activity of tritium.

Thus the Cs-137, SR-90, and I-129 contents of the ECT tank were assumed to be released to the groundwater. A coincidental release of just the RHT tritium contents was also assumed to provide the maximum offsite nuclide concentrations. The assumptions used in determining the activities can be found in Table 15.7-3 with the activities provided in Table 15.7-4.

The radionuclides considered were obtained by comparing the half-life versus the transit time to the Colorado River (~90 years). The resulting concentrations and methods of dispersion can be found in Section 2.4.13.

15.7.3.3 Radiological Consequences. The radiological consequences of this accident are presented in Section 2.4.13. INSERT 15.7-2

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15.7.4 Design Basis Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description. The design basis fuel handling accident is defined as the dropping of a spent fuel assembly during fuel handling, resulting in the rupture of the cladding of the fuel rods in the assembly despite many administrative controls and physical limitations imposed upon fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

During refueling operations, the Normal Containment Purge Subsystem is operating; this system is described in Section 9.4.5. Should a fuel handling accident occur in the Containment, the Reactor Containment Building (RCB) Purge Isolation monitors are capable of identifying that the activity release has occurred and initiating Containment isolation. The function, instrument type, setpoints, safety class, and other pertinent information on the RCB

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The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation under either scenario would have a negligible impact on the isotopic inventory of the liquid waste processing system, as described in Section 11.2, and on the radiological consequences of a LWPS failure.

The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation with either type of steam generator would have a negligible impact on the isotopic inventory of the liquid waste processing system, as described in Section 11.2, and on the radiological consequences of a LWPS failure.

M = mass of primary coolant, g

λ = radiological decay constant, per sec

The release rate from the fuel has been increased by a factor of 500 (over the equilibrium condition release rate) to model the effect of the spike. The iodine appearance rates in the reactor coolant for normal steady-state operation at 1 μCi/g of dose equivalent I-131 and for an assumed accident-initiated iodine spike are given in Table 15.A-6. The iodine appearance rates for the SGTR are given in Table 15.A-7.

15.A.4 The Impact of Extended Burnup Fuel on Source Terms

The source terms presented in Sections 15.A.1 through 15.A.3 are based on an equilibrium fuel cycle using discharge burnup of 33,000 MWD/MTU. The use of extended burnup fuel at STPEGS has been reviewed in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors" (References 11.1-4, 11.1-5) and has been determined to not significantly change the results previously presented in safety analysis reports based on operation to 33,000 MWD/MTU discharge burnup.

The increase of fuel burnup and enrichment up to 5.0 w/o U-235 will not significantly impact the radiological consequences of both LOCA and non-LOCA accidents discussed in Chapter 15 or the control room operator doses presented in Section 6.4. NUREG/CR-5009 predicts a possible 20% increase in the offsite thyroid dose as a result of a Fuel Handling Accident due to an increase in the release fraction of I-131 into the fuel-clad gap for extended burnup fuel. This increase has been previously reviewed for STPEGS and found to be acceptable (Reference 11.1-5). This increase is reflected in the source term used for the fuel handling accident in RCB as presented in Table 15.7-11.

Source terms based on an equilibrium fuel cycle using batch average burnups of 20,000 MWD/MTU, 40,000 MWD/MTU, and 60,000 MWD/MTU (each at 1/3 core size) with fuel enriched to a nominal 5.0 w/o U-235, have been evaluated and do not significantly change the results in sections 15.A.1 through 15.A.3 or the raiological consequences presented in Chapter 15.

15.A.5 Applicability of V5H Fuel Upgrade

The effect of the V5H fuel upgrade on the radioactivity concentrations in the fluid systems was reviewed and it was determined that the original reactor core activity listed in Table 15.A-1 is bounding. Therefore, the UFSAR analyses based on this activity are not adversely impacted by the fuel upgrade. For comparision, the activity concentrations calculated for the V5H fuel are listed in Table 15.A-1A. The corresponding reactor coolant activity for the V5H upgrade is shown in Table 11.1-2A.

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15.A.6 The Impact of Operating at a Reduced Feedwater Temperature on Source Terms

The impact of operating at a feedwater temperature as low as 420 °F for Model E steam generators or as low as 390 °F for Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation under either scenario would have a negligible impact on the fission product inventories in the plant systems. The impact of the changes on the reactor coolant inventory and the inventory in the secondary side was evaluated and determined to have a negligible impact on the activities of these systems.

15.A.7 The Impact of Westinghouse Model Delta 94 Replacement Steam Generators on Source Terms

The impact of replacing the Westinghouse Model E with Westinghouse Model Delta 94 steam generators on the radiological source terms has been evaluated. It was determined that operation with either type of steam generator would have a negligible impact on the fission product inventories in the plant systems. The impact of the changes on the reactor coolant inventory and the inventory in the secondary side was evaluated and determined to have a negligible impact on the activities of these systems.

APPENDIX 15.B

DOSE MODELS

This appendix describes the mathematical models and parameters used for the fission product transport from the postulated accident site to the environment and for the radiological dose calculations.

15.B.1 General Accident Parameters

This section describes the parameters used in analyzing the radiological consequences of postulated accidents. The site-specific, 5-percentile, short-term dispersion factors for the worst sector (assuming ground level releases) are given in Table 15.B-1. (See Section 2.3.4 for additional details on meteorology.) The breathing rates used are presented in Table 15.B-2. The thyroid (via inhalation pathway), beta skin, and gamma body (via submersion pathway) dose factors ~~based upon Reference 15.B-3 are given~~ are discussed in Section 15.B.4 and in Table 15.B-3. IN 2272

15.B.2 Offsite Radiological Consequences Calculational Models

This section presents the models and equations used for calculating the integrated activity released to the environment, the accident flow paths, and the equations for dose calculations. Two major release models are considered:

- A single holdup system with no internal cleanup
- A holdup system wherein a two-region spray model is used for internal cleanup

15.B.2.1 Accident Release Pathways. The release pathways for the major accidents are given in Figure 15.B-2. The accidents and their pathways are as follows:

1. Loss-of-Coolant Accident (LOCA)

   Immediately following a postulated LOCA, the release of radioactivity from the containment is to the environment with the containment spray and Engineered Safety Features (ESF) systems in full operation. The release in this case is calculated using Equations 15.B.2-6 and 15.B.2-7, which take into account a two-region spray model within the Containment. The release of radioactivity to the environment due to assumed ESF system leakages in the Fuel Handling Building (FHB) will be via ESF filters and is calculated using Equation 15.B.2-5.

15.B.3.5 Control Room Gamma Body Dose Calculation. Due to the finite size of the control room, the gamma body doses to a control room operator will be substantially less than what they would be due to immersion in an infinite cloud of gamma emitters. The finite cloud gamma doses are calculated using Murphy's method (Ref. 15.B-2) which models the control room as a hemisphere. The following equation is used:

$$D_{\gamma\text{-}CR} = \frac{1}{GF} \sum_i DCF_{\gamma i} \sum_j (IA_{CRij}) \times o_j \qquad \text{(Eq. 15.B.3-5)}$$

where:

$D_{\gamma\text{-}CR}$ = gamma body dose in the control room, rem

GF = dose reduction due to control room geometry factor

GF = $1173/V_1^{0.338}$, dimensionless

V_1 = volume of the control room, ft^3

$DCF_{\gamma i}$ = gamma body dose conversion factor for isotope i, rem-m^3/Ci-sec

IA_{CRij} = integrated activity concentration in control room, Ci-sec/m^3 for isotope i during time interval j

o_j = control room occupancy fraction during time interval j

15.B.3.5.1 Model for Radiological Consequences Due to Radioactive Cloud External to the Control Room: This dose is calculated based upon the semi-infinite cloud model (Section 15.B.2.6) which is modified by multiplying by a protection factor to account for the control room walls.

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15.B.4 Dose Conversion Factors

The thyroid (via inhalation pathway), beta skin, and gamma body (via submersion pathway) dose factors based upon Regulatory Guide 1.109 (Reference 15.B-3) are given in Table 15.B-3.

For certain analyses, dose conversion factors were derived from ICRP 30 data (Reference 15.B-5) as an alternative to those based on Regulatory Guide 1.109. These DCFs may be used as a replacement for the DCFs based upon Regulatory Guide 1.109 for control room, Technical Support Center (TSC), and offsite calculations. However, unless stated in the accident description, the DCFs based upon Regulatory Guide 1.109 were used in an analysis.

Thyroid DCF
The tabulated ICRP 30-based thyroid DCFs listed in Table 15.B-3 all originate from Federal Guidance Report 11 (Reference 15.B-7). These coefficients give committed dose equivalence (CDE) to the thyroid per unit activity of inhaled radionuclides. The coefficients were calculated using the most recent metabolic and physiologic modeling and should provide the best estimate of thyroid dose.

Skin DCF
The most recent publication for skin dose conversion factors is Federal Guidance Report 12. However, these reported DCF contain contributions to skin dose from both photons and electrons. The skin DCFs are partially corrected for gamma contribution based on the control room volume. This gives a more conservative dose calculation than beta alone. The total skin DCFs were taken from Reference 15.B-7, with the exception of Kr-89 and Xe-137, which were taken from Reference 15.B-8. The "beta skin dose" is analogous to the Shallow Dose Equivalent (SDE) dose.

The larger volume of the control room will also make a conservative gamma correction to the skin DCF for use with the smaller Technical Support Center. This is because the Murphy-Campe (Reference 15.B-2) geometry factor term is inversely proportional to the volume, and the DCF correction is inversely related to the geometry factor, which makes the DCF directly related to the node volume. Therefore, the larger control room volume makes a conservatively larger DCF.

The skin DCFs are conservative to use for offsite doses. This is because the Regulatory Guide 1.109 for skin doses are based on beta exposure only. Including the control room volume-corrected gamma contribution in the offsite skin doses is more conservative than beta only.

Total Body DCF
The Total Body DCF taken from Federal Guidance Report 12 (Reference 15.B-7) assumes submersion in a semi-infinite cloud of effluent. The cloud concentration is assumed to be uniform throughout the problem domain. Whole body DCFs were taken from Reference 15.B-7, with the exception of Kr-89 and Xe-137, which were taken from Reference 15.B-8. The "whole body dose" is equivalent to the Deep Dose Equivalent (DDE) dose.

REFERENCES

Appendix 15.B:

15.B-1 Report of Committee II on Permissible Dose for Internal Radiation, International Commission on Radiation Protection (ICRP) Publication 2 (1959).

15.B-2 Murphy, K.G. and Campe, K.M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" Paper presented at the 13th AEC Air Cleaning Conference.

15.B-3 "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", USNRC Regulatory Guide 1.109, Rev. 1, October 1977.

15.B-4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," USNRC Regulatory Guide 1.4 Rev. 2, June 1974

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15.B-5 International Commission on Radiation Protection (ICRP), "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, *Annals of the ICRP Volume 2*, 1979.

15.B-6 US Environmental Protection Agency, *"Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,"* Federal Guidance Report No. 11, EPA-520/1-88-020, September, 1988.

15.B-7 US Environmental Protection Agency, *"External Exposure to Radionuclides in Air, Water, and Soil,"* Federal Guidance Report No. 12, EPA 402-R-93-081, September 1993.

15.B-8 U.S. Department of Energy, *"External Dose-Rate Conversion Factors for Calculation of Dose to the Public,"* DOE/EH-0070, July, 1988.

TABLE 15.B-3

DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

Nuclide	Total Body rem-m³/Ci-sec	Beta Skin rem-m³/Ci-sec	Thyroid (rem/Ci)
I-131	NA	NA	1.49E+6
I-132	NA	NA	1.43E+4
I-133	NA	NA	2.69E+5
I-134	NA	NA	3.73E+3
I-135	NA	NA	5.60E+4
Kr-83M	2.40E-6	NA	NA
Kr-85M	3.71E-2	4.63E-2	NA
Kr-85	5.1E-4	4.25E-2	NA
Kr-87	1.88E-1	3.08E-1	NA
Kr-88	4.66E-1	7.51E-2	NA
Kr-89	5.26E-1	3.2E-1	NA
Xe-131m	2.9E-3	1.51E-2	NA
Xe-133m	7.96E-3	3.15E-2	NA
Xe-133	9.32E-3	9.70E-3	NA
Xe-135m	9.9E-2	2.25E-2	NA
Xe-135	5.38E-2	5.90E-2	NA
Xe-138	2.80E-1	1.31E-1	NA

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TABLE 15.B-3
DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

	ICRP 2 and Reg Guide 1.109 Based			ICRP 30 - Based		
	Total Body	Beta Skin	Thyroid	Total Body	Beta Skin	Thyroid
Nuclide	(rem-m^3/ ci-sec)	(rem-m^3/ ci-sec)	(rem/ci)	(rem-m^3/ ci-sec)	(rem-m^3/ ci-sec)	(rem/ci)
I-131	8.72E-2	3.17E-2	1.49E+6	6.734E-2	4.087E-2	1.080E+6
I-132	5.13E-1	1.32E-1	1.43E+4	4.144E-1	1.617E-1	6.438E+3
I-133	1.55E-1	7.35E-2	2.69E+5	1.088E-1	1.032E-1	1.798E+5
I-134	5.32E-1	9.23E-2	3.73E+3	4.810E-1	2.011E-1	1.066E+3
I-135	4.21E-1	1.29E-1	5.60E+4	2.953E-1	1.153E-1	3.130E+4
Kr-83M	2.40E-6	NA	NA	5.550E-6	1.547E-5	NA
Kr-85M	3.71E-2	4.63E-2	NA	2.768E-2	5.468E-2	NA
Kr-85	5.1E-4	4.25E-2	NA	4.403E-4	4.843E-2	NA
Kr-87	1.88E-1	3.08E-1	NA	1.524E-1	3.482E-1	NA
Kr-88	4.66E-1	7.51E-2	NA	3.774E-1	1.221E-1	NA
Kr-89	5.26E-1	3.2E-1	NA	3.232E-1	3.981E-1	NA
Xe-131m	2.9E-3	1.51E-2	NA	1.439E-3	1.544E-2	NA
Xe-133m	7.96E-3	3.15E-2	NA	5.069E-3	3.227E-2	NA
Xe-133	9.32E-3	9.70E-3	NA	5.772E-3	1.145E-2	NA
Xe-135m	9.89E-2	2.25E-2	NA	7.548E-2	3.144E-2	NA
Xe-135	5.38E-2	5.90E-2	NA	4.403E-2	7.066E-2	NA
Xe-137	4.50E-2	3.87E-1	NA	3.026E-2	4.642E-1	NA
Xe-138	2.80E-1	1.31E-1	NA	2.135E-1	1.728E-1	NA