Supplement 1
to WCAP-10750

EVALUATION OF OFFSITE RADIATION DOSES FOR A STEAM GENERATOR TUBE RUPTURE ACCIDENT

R. N. Lewis

P. Huang

K. Rubin

May 1985

APPROVED BY: for E. P. Rahe, Jr., Manager
Nuclear Safety Department

Work Performed Under Shop Order MUHN-5000
For the SGTR Subgroup of the Westinghouse Owners Group

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P.O. Box 355
Pittsburgh, Pennsylvania 15230

8480Q:1D/052085

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

The analysis methodology to determine the margin to steam generator overfill for a steam generator tube rupture (SGTR) accident was developed and presented in WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," dated December 1984. The analysis methodology is based on the simulation of the operator actions for SGTR recovery in Revision 1 of the Emergency Response Guidelines which were developed by the Westinghouse Owners Group (WOG). The operator action times required to perform the recovery operations were evaluated and appropriate times were determined for use in the analysis. Sensitivity studies were performed to determine conservative initial conditions and assumptions and the results of these analyses were used to establish the design basis accident to be used for the SGTR analysis.

A reference plant design was established to serve as the basis for application of the analysis methodology. The equipment which is used during recovery from an SGTR was reviewed and the equipment for which credit is assumed to prevent overfill for the reference plant was identified. The potential single failures were identified from the design basis equipment list and an evaluation was performed to determine the worst single failure with respect to the margin to steam generator overfill. [

margin to steam generator overfill.] [a,c] An analysis was performed for the design basis SGTR for the reference plant using the revised methodology and assuming the worst single failure. As indicated in WCAP-10698, the results of this analysis demonstrated that there is margin to steam generator overfill for the reference plant.

Although the demonstration of margin to steam generator overfill is one of the major SGTR concerns, it is also necessary to demonstrate that the offsite radiation exposure for an SGTR where overfill does not occur will be within

the allowable dose guidelines. Therefore, an evaluation was performed to determine the offsite radiation doses for the single failure cases considered in WCAP-10698, and the results are presented in this supplement to WCAP-10698. The results of this evaluation can be used to determine the magnitude of the expected offsite radiation doses for an SGTR and to identify the worst single failure with respect to the offsite doses.

2.0 REFERENCE PLANT AND SITE

The reference plant which was selected in WCAP-10698 for the evaluation of the margin to steam generator overfill is [ ]^{a,c}

The calculation of the mass releases and the evaluation of the offsite radiation doses are based on the results of the single failure analyses performed for the reference plant.

Since the offsite radiation doses are dependent upon site specific parameters, it is necessary to define a reference site to be used as a basis for these calculations. The site specific parameters which directly affect the dose calculations are the atmospheric dispersion factors (χ/Qs). These factors are a function of meteorology and the distance from the plant release point to the site boundary and to the outer boundary of the low population zone. It is necessary to calculate the radiation doses at the site boundary for a 2 hour exposure period and at the low population zone for the duration of the accident. The duration of the accident is assumed to be 8 hours for an SGTR since the radiological releases to the atmosphere will be terminated within 8 hours. For the purposes of the dose calculation, a reference site was established which has accident χ/Qs of [ ]^{a,c} second/meter^3 at the site boundary for the 0-2 hour exposure period and [ ]^{a,c} second/meter^3 at the low population zone for the 0-8 hour exposure period. These χ/Qs are representative of those which have been utilized for accident analyses in the Final Safety Analysis Reports (FSARs) for typical Westinghouse plants. The offsite radiation exposures were then determined for the single failure cases analyzed in WCAP-10698 for the reference plant using the reference site parameters described above.

3.0 SINGLE FAILURE CASES EVALUATED

The single failures which were considered in the evaluation of the margin to steam generator overfill in WCAP-10698 are listed in Table 1. An assessment was made in WCAP-10698 for each of the single failure cases to determine the decrease in the margin to overfill relative to a conservative base case, and the results of this assessment are also shown in Table 1. The decrease in the margin to overfill for each case is due to the increased operator action time and/or system response time required to complete the recovery operation because of the single failure considered. For several of the single failure cases considered, a LOFTTR1 analysis was performed to determine the margin to overfill, whereas a conservative estimate of the additional operator action time and system response time was used to assess the decrease in the margin to overfill for other cases. For the single failure cases which were analyzed with LOFTTR1, the results of the LOFTTR1 analyses have been used to evaluate the mass releases to the atmosphere and the resulting offsite radiation exposure. For the cases in which no LOFTTR1 results are available, it can be shown that the offsite radiation exposure for these cases would be comparable to the results for the cases which were analyzed.

The single failure cases which have been used for the offsite dose evaluation are discussed below.

Case 1: [ ] a,c

Case 2: [ ] a,c

a,c

Case 3:

a,c

Case 4:

a,c

Case 5:

a,c

Case 6: [ ] a,c

For the remaining single failure cases examined in WCAP-10698, the decrease in the margin to steam generator overfill relative to the conservative base case was determined from the estimated increase in the operator action time and system response time due to the failure. In general, the decrease in the margin to overfill represents the additional net accumulation of water in the secondary side of the ruptured steam generator. Net water accumulation and offsite doses both depend upon the mass of primary coolant transferred to the ruptured steam generator and the mass of steam released from the ruptured steam generator. For the remaining single failure cases, the decrease in the margin to overfill is primarily due to the additional reactor coolant transferred to the secondary side of the ruptured steam generator as a result of the increased operator action time and/or the system response time, rather than due to changes in the amount of steam released from the ruptured steam generator. Since the decrease in the margin to overfill for the remaining

single failure cases was less than for the design basis case [ ]a,c the offsite radiation exposures for these cases are expected to also be less than for the design basis case. Hence, it is expected that the results for Case 1 above will bound the radiation doses for the remaining single failure cases for which doses were not specifically evaluated. [

]a,c

The mass releases to the atmosphere were determined for each of the cases identified above for the reference plant, and the resulting offsite radiation doses were calculated for each case based on the reference site. Since these results represent the calculated offsite radiation exposure based on the revised SGTR analysis methodology in WCAP-10698, it is desirable to compare these results with the calculated offsite exposure based on the SGTR analysis methodology previously utilized in FSARs. Therefore, the mass releases which were calculated for the reference plant using the previous FSAR methodology were also determined. Because the dose analysis methodology used previously in some FSARs is different than the current methodology, the offsite radiation doses were calculated for the reference site using the FSAR mass releases and the same dose analysis methodology used to evaluate the above single failure cases. The results of the calculated offsite radiation doses based on the revised SGTR methodology and assumed single failures can then be compared directly with the calculated doses based on the previous FSAR methodology for SGTR analysis.

4.0 CALCULATION OF MASS RELEASES

The mass releases were determined for each of the single failure cases identified in Section 3.0 for use in evaluating the offsite radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours are used to calculate the radiation doses at the site boundary for a 2 hour exposure, and the releases for 0-8 hours are used to calculate the radiation doses at the low population zone for the duration of the accident.

The revised methodology used for the SGTR analysis in WCAP-10698 is based on the simulation of the operator actions for SGTR recovery in the WOG Emergency Response Guidelines, and the results of the LOFTTR1 analyses from WCAP-10698 were used to determine the SGTR mass releases prior to termination of the primary to secondary leakage. Although the LOFTTR1 analyses were only performed until primary to secondary leakage termination, the plant conditions and the mass releases after leakage termination were also determined based on a continuation of the SGTR recovery operations in the Emergency Response Guidelines.

4.1 OPERATOR ACTIONS FOR SGTR RECOVERY

In the LOFTTR1 analyses, the SGTR recovery actions in the E-3 guideline were simulated until the termination of primary to secondary leakage. After the primary to secondary leakage is terminated, the operators will continue the SGTR recovery actions in the E-3 guideline to prepare the plant for cooldown to cold shutdown conditions. These actions include establishing normal Chemical and Volume Control System (CVCS) operation to provide reactor coolant inventory control and a boration path; restarting a reactor coolant pump (RCP), if none are running, to ensure homogeneous RCS conditions and to provide normal pressurizer spray; or stopping all but one RCP, if more than one is running, to minimize the heat input during the subsequent cooldown; and the actions necessary to minimize the spread of contamination on the

secondary side. When the instructions provided in E-3 are completed, the plant should be cooled and depressurized to cold shutdown conditions. There are three alternate means of performing the post-SGTR cooldown provided in the WOG Emergency Response Guidelines. The guidelines are: ES-3.1, POST-SGTR COOLDOWN USING BACKFILL; ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN; and ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP. A decision will have to be made to determine which of the post-SGTR cooldown methods to be used after the completion of the E-3 guideline. The preferred methods are using backfill or blowdown since these methods minimize the radioactivity released to the atmosphere. The ES-3.3 guideline using steam dump provides the fastest method for depressurizing the RCS and ruptured steam generator. This method also results in the worst radiological releases, especially if steam dump to the condenser is unavailable. Therefore, the method using steam dump was selected for evaluation of the long-term mass releases since this produces conservative results for the offsite dose evaluation. It is noted that the use of the steam dump method would not be permitted if steam generator overfill occurs and water enters the main steamlines.

The high level actions for the ES-3.3 guideline are discussed below.

1. Prepare for Cooldown to Cold Shutdown

   The initial steps to prepare for cooldown to cold shutdown are performed in the E-3 guideline following SI termination, and these steps will be continued in ES-3.3 if they have not already been completed. A few additional steps are also performed in ES-3.3 prior to initiating cooldown. These include isolating the cold leg SI accumulators to prevent unnecessary injection, energizing pressurizer heaters as necessary to saturate the pressurizer water and provide for better pressure control, and assuring adequate shutdown margin in the event of potential boron dilution due to in-leakage from the ruptured steam generator.

2. Cooldown RCS to Residual Heat Removal (RHR) System Temperature

The RCS is cooled by steaming and feeding the intact steam generators similar to a normal cooldown. Since all immediate safety concerns have been resolved, the cooldown rate should be maintained less than the maximum allowable rate of 100°F/hr. The preferred means for cooling the RCS is steam dump to the condenser since this minimizes the radiological releases and conserves feedwater supply. The PORVs for the intact steam generators can also be used if steam dump to the condenser is unavailable. When RHR system operating temperature is reached, the cooldown is stopped until RCS pressure can also be decreased. This ensures that pressure/temperature limits will not be exceeded.

3. Depressurize RCS to RHR System Pressure

When the cooldown to RHR system temperature is completed, the pressure in the ruptured steam generator is decreased by releasing steam from the ruptured steam generator. Steam release to the condenser is preferred since this minimizes radiological releases. However, steam can also be released to the atmosphere using the PORV on the ruptured steam generator. An evaluation of the potential radiological consequences should be performed before releasing steam from the ruptured steam generator to the atmosphere. As the ruptured steam generator pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured steam generator in order to prevent in-leakage of secondary side water or additional primary to secondary leakage. Normal pressurizer spray is the preferred means of RCS pressure control since this conserves coolant inventory. If pressurizer spray is not available, a pressurizer PORV or auxiliary spray can be used to control RCS pressure.

4. Cooldown to Cold Shutdown

When RCS temperature and pressure have been reduced to the RHR system in-service values, RHR system cooling is initiated to complete the

cooldown to cold shutdown. When cold shutdown conditions are achieved, the pressurizer can be cooled to terminate the event.

4.2 METHODOLOGY FOR CALCULATION OF MASS RELEASES

[ ]a,c

a,c

4.3 MASS RELEASES

The mass release calculations were performed for the single failure cases discussed in Section 3.0, using the methodology discussed above. For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR1 results for two separate periods for use in the dose calculations. The first time period considered is from accident initiation until reactor trip and the second period is from reactor trip until the time of leakage termination. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip will be through the condenser air ejector. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator PORVs. The mass releases calculated from the time of leakage termination until 2 hours and from 2-8 hours are also assumed to be released to the atmosphere via the steam generator PORVs. The mass releases for the design basis SGTR event [ ]a,c (Case 1) are presented in Table 2. The results indicate that approximately [ ]a,c lbm of steam is released from the ruptured steam generator to the atmosphere in the first 2 hours. An additional [ ]a,c lbm of steam is released to the atmosphere from the ruptured steam generator from 2 to 8 hours. A total of [ ]a,c lbm of primary water is transferred to the secondary side of the ruptured steam generator before the break flow is terminated.

It is also necessary to determine the fraction of the primary break flow which flashes to steam in the ruptured steam generator for the purpose of calculating the coolant activity released to the atmosphere. The break flow flashing fraction during the SGTR transient is presented in Figure 1 for Case 1. [ ]a,c

[ ]a,c

[ ]a,c The differential pressure and the water level[ ]a,c are presented in Figures 2 and 3, respectively for Case 1.

The mass releases were determined for each of the cases considered in Section 3.0, similar to the results presented in Table 2. The flashing fraction, primary to secondary pressure differential and the water level[ ]a,c were also determined as a function of time similar to the results in Figures 1, 2, and 3. The mass releases, feedwater flows, and primary to secondary break flow for each of the cases are summarized in Table 3. The mass releases which were calculated for the reference plant using the previous FSAR methodology for SGTR analysis are also presented as Case 7 in Table 3 for comparison.

5.0 CALCULATION OF OFFSITE RADIATION DOSES

For the calculation of the radiological consequences of an SGTR, it is assumed that the reactor has been operating with a small percent of defective fuel for a sufficient time to establish equilibrium concentrations of radionuclides in the reactor and secondary coolant. Radionuclides from the primary coolant enter the steam generator, via the ruptured tube, and are released to the atmosphere via the turbine condenser air ejector exhaust or through the steam generator PORV or safety valves.

The radioactivity released to the atmosphere, due to an SGTR, depends upon the primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, time dependent break flow flashing fractions, time dependent scrubbing of flashed activity, partitioning of activity between the steam generator liquid and steam, and the mass of fluid discharged to the environment. All of these parameters were conservatively evaluated and the radiological consequences were calculated for each of the cases considered in Section 3.0. The calculational methods and the results of the offsite dose calculations are discussed in the following sections.

5.1 ANALYSIS METHODOLOGY AND ASSUMPTIONS

The major assumptions and parameters used in the calculation of the offsite radiation doses are summarized below.

<u>Source Term Calculations</u>

The concentrations of radionuclides in the primary and secondary system, prior to and following the accident were determined as follows:

1. The equilibrium iodine activity in the reactor coolant is assumed to be based on 1 μCi/gram of Dose Equivalent I-131 as shown in Table 4. The iodine concentrations in the reactor coolant for the dose calculations were then determined for both preaccident and accident initiated iodine spikes as indicated in NRC Standard Review Plan 15.6.3, "Radiological Consequences of Steam Generator Tube Failure", Rev. 2, July 1981.

a. Preaccident Spike - It is assumed that a reactor transient has occurred prior to the SGTR which raises the primary coolant iodine concentration from 1 μCi/gram to 60 μCi/gram of Dose Equivalent I-131.

b. Accident Initiated Spike - It is assumed that the primary system depressurization associated with the SGTR creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1 μCi/gram of Dose Equivalent I-131. The duration of the spike is assumed to be [ ]^{a,c} The iodine appearance rates in the reactor coolant for this case are presented in Table 5.

2. The noble gas activity in the reactor coolant is based on 1 percent fuel defects as shown in Table 4. The assumption of 1 percent fuel defects for the calculation of noble gas activity is conservative, since 1 μCi/gram Dose Equivalent of I-131 and 1 percent defects cannot exist simultaneously. Iodine activity based on 1 percent defects would be greater than twice the Standard Technical Specification limit based on 1 μCi/gram of Dose Equivalent I-131.

3. The secondary coolant activity is based on the Dose Equivalent of 0.1 μCi/gram of I-131 as shown in Table 4.

Dose Calculations

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following a SGTR.

1. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the intact and ruptured steam generators to the atmosphere are presented in Table 3 for each case.

2. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment was determined for each case. Typical results are shown in Figure 1 for Case 1.

3. The time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the leak site [ ]a,c to the water surface was also determined for each case. [ ]a,c The iodine removal efficiency is shown in Figure 4 for Case 1.

4. The 1 gpm primary to secondary leak is assumed to be split evenly between the steam generators.

5. All noble gas activity in the reactor coolant which is transported to the secondary system via the tube rupture and the primary to secondary leakage is assumed to be immediately released to the atmosphere.

6. The iodine partition factor between the liquid and steam of the ruptured and intact steam generators is assumed to be 100.

7. No credit was taken for radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the site boundary or outer boundary of the low population zone.

8. Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table 6. The breathing rates were obtained from NRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors", Rev. 2, June 1974.

9. Decay constants and dose conversion factors for the gamma-body and thyroid doses are presented in Table 7. The dose conversion factors were obtained from NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I", Rev. 1, October 1977.

10. The offsite thyroid doses were calculated using the equation:

$$D_{Th} = \sum_i DCF_i \sum_j (IAR)_{ij} \quad (BR)_j \; (x/Q)_j$$

where:

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j in Ci

$(BR)_j$ = breathing rate during time interval j in meter^3/second

$(x/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

$(DCF)_i$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci

D_{Th} = thyroid dose via inhalation in rems

11. The offsite gamma-body doses were calculated using the following equation, assuming a semi-infinite cloud of gamma emitters.

$$D_{TB} = \sum_i DCF_{\gamma i} \sum (IAR)_{ij} \, (x/Q)_j$$

where

$(IAR)_{ij}$ = Integrated activity of isotope i released during the j^{th} time interval in Ci

$(x/Q)_j$ = offsite atmospheric dispersion factor during time interval j in second/meter^3

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

D_{TB} = gamma-body dose in rems

5.2 OFFSITE RADIATION DOSES

The offsite radiation doses were calculated for each of the cases considered using the analysis methodology and assumptions described above. For Case 1, the thyroid and gamma-body doses were determined at the site boundary for a 2 hour exposure period and at the outer boundary of the low population zone for an 8 hour exposure period, and the results are presented in Table 8. The doses were determined for both a preaccident and an accident initiated iodine spike. Standard Review Plan 15.6.3 indicates that the calculated doses for a preaccident initiated iodine spike should not exceed the guideline values of 10 CFR 100, while for an accident initiated iodine spike, the calculated doses should not exceed a small fraction of the 10 CFR 100 guidelines, i.e., 10 percent of the 10 CFR 100 limits. The guideline values for the two iodine spike conditions are also presented in Table 8 for comparison with the calculated doses.

It can be seen from Table 8 that all of the calculated doses are well within the guidelines, and that the thyroid doses are controlling. In addition, it is noted that the thyroid doses for the accident initiated iodine spike are more limiting than for the preaccident iodine spike relative to the guideline values, although the magnitude of the thyroid doses for the preaccident iodine spike is greater. Based on this information, it is concluded that the thyroid

doses for an accident initiated iodine spike represent the most limiting condition for the evaluation of the offsite radiation exposure for the different SGTR cases considered. This is consistent with previous FSAR evaluations of the offsite radiation doses for a design basis SGTR. Hence, only the thyroid doses for the accident initiated iodine spike were calculated for the remaining cases to evaluate the relative consequences of the different single failures.

The thyroid doses at the site boundary and the low population zone for the accident initiated iodine spike are presented in Table 9 for each of the cases considered. The thyroid doses were also calculated using the mass releases which were determined for the reference plant based on the previous FSAR methodology and the dose analysis methodology described above, and the results are presented as Case 7 in Table 9. The results in Table 9 show that the calculated radiation doses for each of the single failure cases are within the allowable guidelines. [

]a,c

[ ]a,c

a,c

a,c

a,c

Although sensitivity studies were not specifically performed to identify conservative conditions for the offsite dose evaluation, most of the conservative assumptions and initial conditions which were used for the

evaluation of the margin to overfill are also expected to be conservative for the offsite dose evaluation. This follows from the fact that the offsite doses are primarily dependent upon the amount of primary to secondary leakage and steam released from the ruptured steam generator, which are also the most important considerations in determining the potential for overfill. [

]a,c

6.0 DISCUSSION OF RESULTS

An evaluation was performed to determine the offsite radiation doses for a design basis SGTR considering various single failures. The evaluation is based on the results of the single failure analyses which were performed in WCAP-10698 to determine the margin to steam generator overfill. The offsite radiation doses were determined using the mass releases calculated for the reference plant and the meteorological parameters for a reference site. The results of this evaluation show that the offsite doses for the reference plant and reference site will be within the Standard Review Plan guidelines for each of the single failure cases considered. [

]a,c

[

]a,c

TABLE 1

SINGLE FAILURE CASES CONSIDERED FOR EVALUATION
OF THE MARGIN TO STEAM GENERATOR OVERFILL

Equipment Failure	Decrease in Margin to Overfill Relative to Conservative Base Case (Min)
AFW Flow Control Valve	[ ]a,c
Ruptured SG PORV	
MSIV	
Steam Supply Valve for Turbine-driven AFW pump	
Main FW Flow Control Valve	
Emergency Diesel	
Intact SG PORV	
Pressurizer PORV	
SI Pump Switches	
BIT Isolation Valves	

TABLE 2

MASS RELEASES FOR A DESIGN BASIS SGTR ASSUMING LOSS
OF ONE INTACT SG PORV (CASE 1)

	INTEGRATED MASS FLOWS (lbm) FOR EACH TIME PERIOD			
	0 - TTRIP	TTRIP-TTBRK	TTBRK-T2HRS	T2HRS-TRHR
Ruptured SG				a,c
Condenser				
Atmosphere				
Feedwater				
Intact SG				
Condenser				
Atmosphere				
Feedwater				
Break Flow				

TTRIP = Time of reactor trip = [ ]a,c
TTBRK = Time to terminate break flow = [ ]a,c
T2HRS = Time at 2 hours = 7200 sec
TRHR = Time to reach RHR in-service conditions = 28,800 seconds

TABLE 3

SUMMARY OF SGTR MASS RELEASES

	INTREGATED MASS FLOWS (lbm)								
	Case 1	Case 2a	Case 2b	Case 2c	Case 3	Case 4	Case 5	Case 6	Case 7*
Ruptured SG									
To Condenser									
0 - 2 hr									
2 - 8 hr									
To Atmosphere									
0 - 2 hr									
2 - 8 hr									
Intact SG									
To Condenser									
0 - 2 hr									
2 - 8 hr									
To Atmosphere									
0 - 2 hr									
2 - 8 hr									
Primary to Secondary Leakage									

a,c

*The mass releases for this case were calculated for the reference plant using the previous FSAR methodology for SGTR analysis.

TABLE 4

REACTOR COOLANT IODINE AND NOBLE GAS ACTIVITY

Nuclide	*Iodine Activity based on 1 μCi/gram of Dose Equivalent I-131 (μCi/gram)
I-131	0.76
I-132	0.27
I-133	1.16
I-134	0.16
I-135	0.64
	Noble Gas Activity Based on 1 percent Fuel Defects (μCi/gram)
Xe-131m	2.1
Xe-133m	17.0
Xe-133	260
Xe-135m	0.5
Xe-135	7.2
Xe-138	0.7
Kr-85m	2.0
Kr-85	7.7
Kr-87	1.3
Kr-88	3.7

*Secondary coolant iodine activity is based on 0.1 μCi/gram of Dose Equivalent I-131 and is therefore 10 percent of these values.

TABLE 5

IODINE APPEARANCE RATES IN THE
REACTOR COOLANT FOR A DESIGN BASIS SGTR
(CURIES/SECOND)

	I-131	I-132	I-133	I-134	I-135
Equilibrium Appearance Rates due to Technical Specification Fuel defects	3.3×10^{-3}	5.0×10^{-3}	6.8×10^{-3}	6.8×10^{-3}	6.0×10^{-3}
Appearance Rates due to an Iodine Spike-500X equilibrium rates	1.7	2.6	3.4	3.4	3.0

TABLE 6

SHORT-TERM ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES
FOR ACCIDENT ANALYSIS

Time (hours)	Site Boundary χ/Q (sec/m^3)	Low Population Zone χ/Q (sec/m^3)	Breathing Rate (m^3/sec)
0-2	[ ]^{a,c}	-	3.47×10^{-4}
0-8	-	[ ]^{a,c}	3.47×10^{-4}

TABLE 7

ISOTOPIC DATA

Isotope	Decay Constant (1/Hr)	Dose Conversion Factors: Gamma-Body $rem\text{-}m^3/Ci\text{-}sec$	Dose Conversion Factors: Thyroid (rem/Ci)
I-131	0.00359	-	1.49(6)
I-132	0.301	-	1.43(4)
I-133	0.033	-	2.69(5)
I-134	0.800	-	3.73(3)
I-135	0.103	-	
Xe-131m	0.00245	2.91(-3)	-
Xe-133m	0.0128	7.97(-3)	-
Xe-133	0.00548	9.33(-3)	-
Xe-135m	2.67	9.91(-2)	-
Xe-135	0.0753	5.75(-2)	-
Xe-138	2.45	2.80(-1)	-
Kr-85m	0.158	3.71(-2)	-
Kr-85	0.00000735	5.11(-4)	-
Kr-87	0.547	1.88(-1)	-
Kr-88	0.248	4.67(-1)	-

TABLE 8

CALCULATED OFFSITE RADIATION DOSES FOR A DESIGN BASIS SGTR
ASSUMING LOSS OF ONE INTACT SG PORV (CASE 1)

	Doses (Rem)			
	Site Boundary (0-2 hr)		Low Pop. Zone (0-8 hr)	
	Thyroid	Total Body	Thyroid	Total Body
1. Pre-Accident Iodine Spike				
Calculated Dose	[			] a,c
Standard Review Plan Guidelines	300	25	300	25
2. Accident Initiated Iodine Spike				
Calculated Dose	[			] a,c
Standard Review Plan Guidelines	30	2.5	30	2.5

TABLE 9

COMPARISON OF CALCULATED THYROID DOSES
FOR AN ACCIDENT INITIATED IODINE SPIKE

	Thyroid Doses (Rem)		
Case	Site Boundary (0-2 hrs)	Low Population Zone (0-8 hrs)	
1			a,c
2a			
2b			
2c			
3			
4			
5			
6			
7*			

* The thyroid doses for this case were calculated using the mass releases which were determined for the reference plant based on the previous FSAR methodology, and the dose analysis methodology described in this report.

FIGURE 1

BREAK FLOW FLASHING FRACTION FOR CASE 1

a,c

FIGURE 2

PRESSURE DIFFERENTIAL BETWEEN RCS AND RUPTURED SG FOR CASE 1

a,c

FIGURE 3

RUPTURED STEAM GENERATOR WATER LEVEL [ ]a,c FOR CASE 1

[ ]a,c

FIGURE 4

IODINE REMOVAL EFFICIENCY

BY BUBBLE SCRUBBING FOR CASE 1

a,c