WCAP-10885

ANALYSIS OF POTENTIAL RADIOLOGICAL CONSEQUENCES FOLLOWING A STEAM GENERATOR TUBE RUPTURE AT THE R. E. GINNA NUCLEAR POWER PLANT USING LOFTTR1

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I. INTRODUCTION

An evaluation of the potential radiological consequences due to a steam generator tube rupture (SGTR) event has been performed for the R. E. Ginna nuclear power plant to demonstrate that the offsite radiation doses will be less than the allowable guidelines based on the Standard Technical Specification limit on primary coolant activity.

A design basis steam generator tube rupture was analyzed for Ginna using the methodology developed in WCAP-10698 (reference 1) and the supplement to WCAP-10698 (reference 2). Two single failure cases were considered to determine which is the most limiting single failure for Ginna with respect to radiological consequences. The two cases examined were:

a,c

Case 1

Case 2

Plant response to the event was modeled using the LOFTTR1 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the faulted steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a steam generator tube rupture based on the Westinghouse Owners Group Emergency Response Guidelines, which are the basis for the Ginna Emergency Operating Procedures. The mass releases were calculated with the LOFTTR1 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases and feedwater flows from the intact and faulted steam generators were determined from a mass and energy balance using the calculated RCS and steam generator conditions at the time of leakage termination. The mass releases for both cases were used to determine the radiation doses at the exclusion area boundary and low population zone assuming that the primary coolant activity is at the Standard Technical Specification limit prior to the accident.

II. THERMAL HYDRAULIC ANALYSIS

Integrated mass releases to the atmosphere and condenser during a steam generator tube rupture event were calculated for various time periods during the accident. This section includes the methods and assumptions used to model the SGTR event and calculate the mass releases, as well as the sequence of events for the recovery.

A. Design Basis Accident

The accident modeled is the complete severance of a steam generator tube located at the tube sheet on the cold leg side. It was also assumed that loss of offsite power occurred at the time of reactor trip, and the worst rod was assumed to be stuck at reactor trip.

a,c

a,c

B. Conservative Assumptions

Plant responses and mass releases from the intact and faulted steam generator prior to break flow termination were calculated using LOFTTR1. While modeling the SGTR event the following assumptions were made:

1. Reactor Trip on Overtemperature delta-T

a,c

2. Power

a,c

3. Pressurizer Water Level

a,c

4. Steam Generator Secondary Mass

a,c

5. Break Location

a,c

6. Reactor Trip Delay

7. Turbine Trip Delay

8. Steam Generator Relief Valve Pressure Setpoint

a,c

9. Pressurizer Pressure for SI Initiation

a,c

10. Leakage after Overfill

a,c

11. Flashing Fraction

a,c

C. Operator Action Times

In the event of an SGTR, the operator is required to take actions to stabilize the plant and terminate the primary to secondary leakage. An evaluation has been performed (reference 1) to establish the operator action times for use in the analysis of a design basis SGTR event. The operator actions which are required for recovery from an SGTR and the available data on the times to perform these actions have been reviewed. The available data on operator action times for an SGTR includes information which has been obtained from reactor plant simulator studies as well as plant data from five actual SGTR events. Using this data, operator action times have been established which are considered to be appropriate for a design basis SGTR event. These operator action times will be used as input for the analysis of the design basis SGTR event.

The major operator action for SGTR recovery which are included in the E-3 guideline of the Westinghouse Owners Group Emergency Response Guidelines were explicitly modeled in the analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator (step 2).

High secondary side activity, as indicated by the steam generator blowdown line radiation monitor or air ejector radiation monitor, typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by high activity in the corresponding steam generator blowdown line, main steamline, or water sample. For an SGTR that results in a high power reactor trip, the steam generator water level

will decrease off-scale on the narrow range for both steam generators. The auxiliary feedwater (AFW) flow will begin to refill the steam generators, typically distributing approximately equal flow to both steam generators. Since primary to secondary leakage adds additional inventory which accumulates in the ruptured steam generator, level will return to the narrow range in that steam generator significantly earlier and will continue to increase more rapidly. This response provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generator and isolate feedwater to the ruptured steam generator.(steps 3 and 4).

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage. In the guideline for steam generator tube rupture recovery in the ERGs, the operator is directed to maintain the level in the ruptured steam generator between just on span and 50% on the narrow range instrument. Reference 1 assumed that the ruptured steam generator would be isolated when level in the steam generator reached midway between these points. For Ginna it was conservative to use 33 percent of level for isolation. If the time to reach 33 percent narrow range level was less than 10 minutes, then 10 minutes was used as the isolation time. The ruptured steam generator was assumed to be isolated at 33 percent narrow range level or at 10 minutes, whichever was longer.

3. Cool down the Reactor Coolant System (RCS) using the intact steam generator (step 14).

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than saturation at the ruptured steam

generator pressure by dumping steam from only the intact steam generator. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. With offsite power available, the normal steam dump system to the condenser will provide sufficient capacity to perform this cooldown rapidly. If offsite power is lost, the RCS is cooled using the power-operated relief valve (PORV) on the intact steam generator since neither the steam dump valves nor the condenser would be available. It is noted that RCS pressure will decrease during the cooldown as shrinkage of the reactor coolant expands the steam bubble in the pressurizer.

4. Depressurize the RCS to restore reactor coolant inventory (steps 17 or 18).

   When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator. To reduce break flow and establish sufficient pressurizer level, RCS pressure is decreased by opening the pressurizer PORV.

5. Terminate SI to stop primary to secondary leakage (steps 21-23).

   The previous actions will have established adequate RCS subcooling, secondary side heat sink, and reactor coolant inventory following an SGTR to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS pressure and

ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator.

Since these major recovery actions will be modelled in the SGTR analysis, it is necessary to establish the times required to perform these actions. Although the intermediate steps between the major actions will not be explicitly modelled, it is also necessary to account for the time required to perform the steps. It is noted that the total time required to complete the recovery operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (i.e., RCS cooldown, RCS depressurization, etc.) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times to initiate RCS cooldown, RCS depressurization and safety injection termination were developed in reference 1 and are listed in Table II.1. In addition to the operator action times developed in reference 1, Rochester Gas and Electric supplied the operator action times associated with recovering from the single failures (Reference 3). [

a,c
]

The times associated with performing these operator actions are listed in Table II.2. [

a,c
]

TABLE II.1

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

a,c

Action	
Identify and isolate ruptured SG	
Operator action time to initiate cooldown	
Cooldown	
Operator action time to initiate depressurization	
Depressurization	
Operator action time to initiate SI termination	
SI termination and pressure equalization	

TABLE II.2

GINNA SPECIFIC OPERATOR ACTION TIMES

a,c

D. Transient Description - Case 1

[

a,c
]

The sequence of events for Case 1 is presented in Table II.3.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure II.4. The RCS pressure also decreases as shown in Figure II.1 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on a overtemperature delta-T setpoint.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs lift to dissipate the energy, as shown in Figure II.3.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the leak flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal. Pressurizer level also decreases more rapidly following reactor trip and the pressurizer eventually empties as shown in Figure II.4. After the pressurizer empties, the RCS pressure rapidly decreases as shown in Figure II.1.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature rise across the core decreases as

core power decays (see Figure II.2), however, the temperature rise subsequently increases as natural circulation flow develops. The cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The RCS temperatures continue to slowly decrease due to the continued addition of the auxiliary feedwater to the steam generators until operator actions are initiated to cool down the RCS.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from the ruptured steam generator and throttling the auxiliary feedwater flow to the ruptured steam generator. As indicated previously the ruptured steam generator is assumed to be identified and isolated when the narrow range level reaches [ ]^{a,c} on the ruptured steam generator or at [ ]^{a,c} minutes after initiation of the SGTR, whichever is longer. For Ginna, the time to reach [ ]^{a,c} is less than [ ]^{a,c} minutes, thus the ruptured steam generator is assumed to be isolated at [ ]^{a,c} minutes.

2. Cool down the RCS to Establish Subcooling Margin

   After isolation of the ruptured steam generator, there is a [ ]^{a,c} minute operator action time imposed prior to cooldown. The actual delay time used in the analysis is 4 seconds longer because of the computer program requirements for simulating the operator actions. After this time, the RCS is cooled as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the PORV on the intact steam generator. [ ]^{a,c}

The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 17°F for subcooling uncertainty. [

a,c

] This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure II.3, and the effect of the cooldown on the RCS temperature is shown in Figure II.2. The RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure II.1.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, a [ ]a,c minute operator action time is included prior to depressurization. The RCS is depressurized at 2542 seconds to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The depressurization is continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 0% plus an allowance of 3% for pressurizer level uncertainty, or pressurizer level is greater than 80% minus an allowance of 3% for pressurizer level uncertainty, or RCS subcooling is less than the 17°F allowance for subcooling uncertainty. The RCS depressurization reduces the break flow as shown in Figure II.6 and increases SI flow to refill the pressurizer, as shown in Figure II.4.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions should have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory following an SGTR to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be

stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated when the RCS pressure increases, minimum AFW flow is available and at least one intact steam generator level is in the narrow range, RCS subcooling is greater than the 17°F allowance for subcooling uncertainty, and the pressurizer level is greater than the 3% allowance for pressurizer level uncertainty. To assure that the RCS pressure is increasing, SI was not terminated in the analysis until the RCS pressure increased to 50 psi above the ruptured steam generator pressure.

After depressurization is completed, an operator action time of [ ] a,c minute is imposed prior to SI termination. After SI is terminated, break flow continues to accumulate in the secondary side resulting in overfill of the ruptured steam generator at 3336 seconds. For the period after overfill occurs, the amount of water released to the atmosphere via the ruptured steam generator PORV is considered equal to the break flow. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generators equalize. This occurs when the intact steam generator PORV is locally opened to cooldown the RCS so that subcooling may be maintained. When the PORV is opened the increased energy transfer from primary to secondary depressurizes the RCS to the ruptured steam generator pressure.

WESTINGHOUSE CLASS 3

TABLE II.3

SEQUENCE OF EVENTS

CASE 1

a,c

EVENT	Time (sec)
Reactor Trip	
Ruptured SG Isolated	
Intact SG PORV Opened	
Intact SG PORV Isolated	
PRZR PORV Opened	
PRZR PORV Closed	
SI Terminated	
Overfill Ruptured SG	
Intact PORV Opened	
Break Flow Terminated	

a,c

CORE PRESSURE - CASE 1

FIGURE II.1

a,c

INTACT LOOP HOT AND COLD LEG RCS TEMPERATURES

CASE 1

FIGURE II.2

a,c

LOOPS A AND B SECONDARY PRESSURE - CASE 1

FIGURE II.3

a,c

PRESSURIZER LEVEL - CASE 1

FIGURE II.4

a,c

RUPTURED SG WATER VOLUME - CASE 1

FIGURE II.5

a,c

PRIMARY TO SECONDARY LEAKAGE - CASE 1

FIGURE II.6

E. Transient Description - Case 2

[

]a,c Thus, the Case 2 transient is identical to the Case 1 transient until that time. The sequence of events for Case 2 is presented in Table II.4.

Following the tube rupture RCS pressure decreases as shown in Figure II.7 due to the primary to secondary leakage. In response to this depressurization, the reactor trips on overtemperature delta-T. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. The steam dump system is inoperable due to the assumed loss of offsite power, which results in the secondary pressure rising to the steam generator PORV setpoint as shown in Figure II.9. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure. Pressurizer level also decreases more rapidly following reactor trip until it eventually empties, as shown in Figure II.10.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   As with Case 1 the ruptured steam generator is assumed isolated at minutes. [

   ]a,c

2. Cool Down the RCS to establish Subcooling Margin

[

a,c
]

The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power is lost the RCS is cooled by dumping steam to the atmosphere using the intact steam generator PORV. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 17°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. For Case 2 cooldown begins at 1806 seconds and is completed at 2626 seconds.

The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure II.9, and the effect of the cooldown on the RCS temperature is shown in Figure II.8. The RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure II.7.

3. Depressurize to Restore Inventory

After the RCS cooldown, a [ ]a,c minute operator action time is included prior to depressurization. The RCS is depressurized at 2748 seconds to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The depressurization is continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 0% plus an allowance of 3% for pressurizer level uncertainty, or pressurizer level is greater than 80% minus an allowance of 3% for pressurizer level uncertainty, or RCS subcooling is less than the 17°F allowance for subcooling

uncertainty. The RCS depressurization reduces the break flow as shown in Figure II.12 and increases SI flow to refill the pressurizer, as shown in Figure II.10.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions should have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory following an SGTR to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated when the RCS pressure increases, minimum AFW flow is available and at least one intact steam generator level is in the narrow range, RCS subcooling is greater than the 17°F allowance for subcooling uncertainty, and the pressurizer level is greater than the 3% allowance for pressurizer level uncertainty. To assure that the RCS pressure is increasing, SI was not terminated until the RCS pressure increased to 50 psi above the ruptured steam generator pressure.

After depressurization is completed, an operator action time of [ ]^{a,c} minute is imposed prior to SI termination. Figure II.12 shows that the primary to secondary leakage continues after the SI flow is stopped until the RCS and ruptured steam generator pressure equalize. For Case 2, the ruptured steam generator does not overfill.

WESTINGHOUSE CLASS 3

TABLE II.4

SEQUENCE OF EVENTS

CASE 2

EVENT	TIME (sec) a,c
Reactor Trip	
Ruptured SG Isolated	
a,c	
Intact SG PORV Opened	
Intact SG PORV Closed	
PRZR PORV Opened	
PRZR PORV Closed	
SI Terminated	
Break Flow Terminated	

a,c

CORE PRESSURE - CASE 2
FIGURE II.7

a,c

INTACT LOOP HOT AND COLD LEG RCS TEMPERATURES

CASE 2

FIGURE II.8

a,c

LOOPS A AND B SECONDARY PRESSURE - CASE 2

FIGURE II.9

a,c

PRESSURIZER LEVEL - CASE 2

FIGURE II.10

a,c

RUPTURED SG WATER VOLUME - CASE 2

FIGURE II.11

a,c

PRIMARY TO SECONDARY LEAKAGE - CASE 2

FIGURE II.12

F. Mass Releases

The mass releases were determined for each of the single failure cases for use in evaluating the exclusion area boundary and low population zone 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 exclusion area 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.

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 one RCP, if both are 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. 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 to 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 generator 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 PORV for the intact steam generator can also be used if steam dump to the condenser is unavailable. When the 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.

When overfill of the ruptured steam generator occurs, as with Case 1, guideline ES-3.1 POST-SGTR COOLDOWN USING BACKFILL is assumed to be used. The high level actions for ES-3.1 are similar to ES-3.3. However, the method by which ES-3.1 instructs the operator to depressurize the ruptured steam generator differs from ES-3.3. In Guideline ES-3.1 the RCS is depressurized to promote back flow through the failed tube which depressurizes the ruptured steam generator without steam releases to the atmosphere.

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.

F.1 Methodology For Calculation Of Mass Releases

a,c

a,c

F.2 Mass Release Results

The mass release calculations were performed for both single failure cases 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. 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 SGTR event [ ]^{a,c} (Case 1) are presented in Table II.5. The results indicate that approximately [ ]^{a,c} lbm of steam and [ ]^{a,c} lbm of water is released from the ruptured steam generator to the atmosphere in the first 2 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.

The mass releases for the SGTR event assuming [ ]^{a,c} (Case 2) are presented in Table II.6. The results indicate that approximately [ ]^{a,c} lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours [ ]^{a,c} lbm is released to the atmosphere from the ruptured steam generator. A total of [ ]^{a,c} lbm of primary water is transferred to the secondary side of the ruptured steam generator before break flow is terminated.

TABLE II.5

CASE 1 MASS RELEASES

TIME PERIOD

	0-TRIP	TRIP - TMSEP	TMSEP - OVFILL	OVFILL - TTBRK	TTBRK - T2HRS	T2HRS - TRHR
Faulted SG						a,c
- Condenser						
- Atmosphere						
- Feedwater						
Intact SG						
- Condenser						
- Atmosphere						
- Feedwater						
Break Flow						

TRIP = Time of reactor trip = [ ] a,c sec.
TMSEP = Time when water reaches the moisture separators = [ ] a,c sec.
OVFILL = Time when steam generator overfills = [ ] a,c sec.
TTBRK = Time when break flow is terminated = [ ] a,c sec.
T2HRS = Time at 2 hours = 7200 sec.
TRHR = Time to reach RHR in-service conditions, 8 hours = 28,800 sec.

TABLE II.6

CASE 2 MASS RELEASES

TOTAL MASS FLOW (POUNDS)

TIME PERIOD

	0-TRIP	TRIP - TMSEP	TMSEP - TTBRK	TTBRK - T2HRS	T2HRS - TRHR
Faulted SG					
- Condenser					
- Atmosphere					
- Feedwater					
Intact SG					
- Condenser					
- Atmosphere					
- Feedwater					
Break Flow					

a,c

TRIP = Time of reactor trip = [ ] sec. a,c
TMSEP = Time when water reaches the moisture separators = [ ] sec. a,c
TTBRK = Time when break flow is terminated = [ ] sec. a,c
T2HRS = Time at 2 hours = 7200 sec.
TRHR = Time to reach RHR in-service conditions, 8 hours = 28,800 sec.

TABLE II.7

SUMMARIZED MASS RELEASES

TOTAL MASS FLOW (POUNDS)

	CASE 1			CASE 2		
	0 - TTBRK	TTBRK - 2HRS	2HRS - 8HRS	0 - TTBRK	TTBRK - 2HRS	2HRS 8HRS
Faulted SG						
- Condenser	47,800	0	0	47,800	0	0
- Atmosphere	37,600	0	0	89,500	17,700	0
- Feedwater	76,400	0	0	91,300	0	0
Intact SG						
- Condenser	47,200	0	0	47,200	0	0
- Atmosphere	79,200	148,800	566,400	67,600	174,200	444,500
- Feedwater	160,900	155,200	575,000	152,400	179,100	444,500
Break Flow	121,000	0	0	166,400	0	0

III. RADIOLOGICAL CONSEQUENCES ANALYSIS

The evaluation of the radiological consequences of a steam generator tube rupture, assumes that the reactor has been operating at the proposed Technical Specification limit for primary coolant activity and at the existing Technical Specification limit for primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator, via the ruptured tube, and are released to the atmosphere through the steam generator safety or power operated relief valves and via the condenser air ejector exhaust.

The quantity of radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated for a design basis double ended rupture of a single tube.

A. Design Basis Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table III.1. The following is a discussion of the source term.

Source Term Calculations

The radionuclide concentrations in the primary and secondary system, prior to and following the SGTR are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.

i. Accident Initiated Spike - The initial primary coolant iodine concentration is 1 μCi/gm of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The duration of the spike, [ ]^{a,c} hours, is sufficient to increase the initial RCS I-131 inventory by a factor of [ ]^{a,c}.

ii. Preaccident Spike - A reactor transient has occured prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 μCi/gram of D.E. I-131.

b. The initial secondary coolant iodine concentration is 0.1 μCi/gram of D.E. I-131.

c. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

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 II.5 and II.6.

2. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is presented in Figure III.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 efficiencies are shown in Figure III.2.

4. The 0.2 gpm primary to secondary leak is assumed to be split evenly between the steam generators.

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

6. 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.

7. Short-term atmospheric dispersion factors (x/Qs) for accident analysis and breathing rates are provided in Table III.4. 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.

Offsite Thyroid Dose Calculation Model

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} \quad (BR)_j \quad (\chi/Q)_j \right) \right]$$

where

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

* No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

$(BR)_j$ = breathing rate during time interval j in meter^3/second (Table III.4)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j in second/meter^3 (Table III.4)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci (Table III.5)

D_{Th} = thyroid dose via inhalation in rem

Results

Thyroid doses at the Exclusion Area Boundary and Low Population Zone are presented in Table III.6. All doses are well within the guidelines of 10CFR100.

TABLE III.1

PARAMETERS USED IN EVALUATING
THE RADIOLOGICAL CONSEQUENCES OF
A STEAM GENERATOR TUBE RUPTURE

I. Source Data	
A. Core power level, MWt	1520
B. Total steam generator tube leakage, prior to accident, gpm	0.2
C. Reactor coolant iodine activity:	
1. Accident Initiated Spike	The initial RC iodine activities based on 1 μCi/gram of D.E. I-131 are presented in Table III.3. The iodine appearance rates assumed for the accident initiated spike are presented in Table III.2.
2. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gram of D.E. I-131 are presented in Table III.3.
D. Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table III.3.
E. Reactor coolant mass, grams	1.27×10^{8}

TABLE III.1 (Sheet 2)

F. Steam generator mass (each), grams	3.39×10^7
G. Offsite power	Lost at time of reactor trip
H. Primary-to-secondary leakage duration for intact SG, hrs.	8
I. Species of iodine	100 percent elemental
II. Activity Release Data	
A. Faulted steam generator	
1. Rupture flow	See Table II.5 or II.6
2. Rupture flow flashing fraction	See Figure III.1
3. Iodine scrubbing plus moisture separator removal efficiency	See Figure III.2
4. Total steam release, lbs	See Table II.5 or II.6
5. Iodine partition factor	
a. Prior to overfill	100
b. After overfill	1.0 - See Figure III.3
6. Location of tube rupture	[ ]^{a,c}

TABLE III.1 (Sheet 3)

B. Intact steam generator	
1. Primary-to-secondary leakage, gpm	0.1
2. Total steam release, lbs	See Table II.5 or II.6
3. Iodine partition factor	100
C. Condenser	
1. Iodine partition factor	100
D. Atmospheric Dispersion Factors	See Table III.4

TABLE III.2

IODINE SPIKE APPEARANCE RATES
(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
0.94	2.22	1.74	3.07	2.34

TABLE III.3

IODINE SPECIFIC ACTIVITIES IN (μCi/gm) THE PRIMARY AND SECONDARY COOLANT BASED ON 1, 60 AND 0.1 μCi/gram OF D.E. I-131

	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.79	47.1	0.079
I-132	0.35	20.7	0.035
I-133	1.01	60.7	0.101
I-134	0.20	12.2	0.020
I-135	0.79	47.1	0.079

TABLE III.4

ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES

Time (hours)	Exclusion Area Boundary χ/Q (Sec/m^3)	Low Population Zone χ/Q (Sec/m^3)	Breathing Rate (m^3/Sec) [4]
0-2	4.8×10^{-4}	3×10^{-5}	3.47×10^{-4}
2-8	-	3×10^{-5}	3.47×10^{-4}

TABLE III.5

THYROID DOSE CONVERSION FACTORS
(Rem/Curie) [5]

Nuclide	
I-131	1.49×10^6
I-132	1.43×10^4
I-133	2.69×10^5
I-134	3.73×10^3
I-135	5.60×10^4

TABLE III.6

RESULTS

	Doses (Rem) Case 1	Case 2
1. Accident Initiated Iodine Spike		
Exclusion Area Boundary (0-2 hr.)	a,c	
Thyroid		4.9
Low Population Zone (0-8 hr.)		
Thyroid		0.3
2. Pre-Accident Iodine Spike		
Exclusion Area Boundary (0-2 hr.)		
Thyroid		30.3
Low Population Zone (0-8 hr.)		
Thyroid		1.9

a,c

BREAK FLOW FLASHING FRACTION - CASE 1

a,c

BREAK FLOW FLASHING FRACTION - CASE 2

FIGURE III.1

a,c

a,c

SCRUBBING AND SEPARATER REMOVAL EFFICIENCY

FIGURE III.2

a,c

a,c

FAULTED SG IODINE PARTITION FACTORS

FIGURE III.3

IV. CONCLUSION

The potential radiological consequences of a steam generator tube failure were evaluated for the R.E. Ginna nuclear power plant to demonstrate that the use of the Standard Technical Specification (STS) primary coolant activity limit of 1 μCi/gram of dose equivalent I-131 will result in offsite doses that are within the appropriate guidelines. The mass releases for a design basis double ended rupture of a single tube with a loss of offsite power were conservatively calculated using the computer code LOFTTR1. Two cases were considered: [

a,c

] The analysis explicitly modeled the time needed for the operators to perform the recovery steps outlined in guideline E-3 of Revision 1 of the Westinghouse Owners Group Emergency Response Guidelines. The resulting doses at the exclusion area boundary and low population zone are within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100. Consequently, the STS primary coolant activity limit is sufficiently low to ensure that the radiological consequences of a steam generator tube rupture at the R.E. Ginna plant will be within the guidelines.

V. REFERENCES

1. Lewis, Volpehein, Huang, Behnke, Fittante, Gelman, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10750 and WCAP-10698, December 1984.

2. Lewis, Huang, Rubin, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10750 and WCAP-10698, May 1985.

3. R. Elaisz, Letter from RGE to Westinghouse concerning Ginna specific operator action times for SGTR analysis, February 7, 1985.

4. NRC Regulatory Guide 1.4, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors", June 1974.

5. NRC Regulatory Guide 1.109, Rev. 1, "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", October 1977.