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August 27, 2014
RC-14-0112

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST – LAR-12-04269
LICENSE BASIS CHANGES IN STEAM GENERATOR
TUBE RUPTURE ANALYSIS

Dear Sir / Madam:

Pursuant to 10 CFR 50.90, South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests prior NRC review and approval to change the Virgil C. Summer Nuclear Station (VCSNS) licensing basis to incorporate supplemental analyses of a steam generator tube rupture (SGTR) accident which explicitly models operator responses and quantifies their impact on the potential for steam generator overfill and offsite and control room doses. The new transient calculations supplement the VCSNS licensing basis analysis by demonstrating margin to steam generator overfill and providing input to a dose analysis that confirms the licensing basis mass transfer input is conservative.

These proposed changes have been reviewed and approved by both the Plant Safety Review Committee and the Nuclear Safety Review Committee. SCE&G has evaluated the proposed changes in accordance with 10 CFR 50.91(a) using criteria in 10 CFR 50.92(c) and has determined that the proposed changes do not involve a significant hazards consideration. SCE&G has also determined that the proposed changes satisfy the criteria for a categorical exclusion in accordance with 10 CFR 51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Although this request is neither exigent nor emergency, your review and approval is requested by August 27, 2015. Once approved, the amendment shall be implemented within 120 days.

A new regulatory commitment to revise the VCSNS Final Safety Analysis Report to reflect the updated SGTR analyses can be found in Attachment 3 of this submittal.

In accordance with 10 CFR 50.91, SCE&G is notifying the State of South Carolina of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Bruce L. Thompson at (803) 931-5042.

A001
NRR

I certify under penalty of perjury that the foregoing is correct and true.

8/27/14 — Executed on

Thomas D. Gatlin

TS/TDG/pr

Enclosure:	Evaluation of the Proposed Change
Attachment 1:	Plant Specific Input to Support Use of WCAP-10698-A
Attachment 2:	Impact Assessment for Fuel Thermal Conductivity Degradation
Attachment 3:	List of Regulatory Commitments

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PRSF RC-14-0112

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12

ENCLOSURE

EVALUATION OF SUPPLEMENTAL STEAM GENERATOR TUBE RUPTURE ANALYSES FOR VCSNS

ENCLOSURE
Evaluation of the Proposed Change

Subject: Supplemental SGTR Analyses for Virgil C. Summer Nuclear Station (VCSNS)

ATTACHMENTS:

1. Plant Specific Input to Support Use of WCAP-10698-P-A

2. Impact Assessment for Fuel Thermal Conductivity Degradation

3. List of Regulatory Commitments

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to update Virgil C. Summer Nuclear Station's (VCSNS) licensing basis to incorporate supplemental analyses of a Steam Generator Tube Rupture (SGTR) accident which explicitly models operator responses and quantifies their impact on the potential for steam generator overfill and offsite and control room doses. The new transient calculations supplement the VCSNS licensing basis analysis by demonstrating margin to steam generator overfill and providing input to a dose analysis that confirms the licensing basis mass transfer input is conservative.

2.0 DETAILED DESCRIPTION

The new SGTR transient analyses are performed by Westinghouse using the LOFTTR2 computer program following the methodology developed in WCAP-10698-P-A and its Supplement 1, with the exception that a single failure was not considered, and with modifications to address NSAL-07-11. The methodology used in WCAP-10698-P-A has been reviewed and accepted by the Nuclear Regulatory Commission (NRC) in safety evaluation report, "Acceptance for Referencing of Licensing Topical Report WCAP-10698, 'SGTR Analysis Methodology to Determine Margin to Steam Generator Overfill,' December 1984," dated March 30, 1987.

New SGTR dose analyses are also performed by Westinghouse using, consistent with the Virgil C. Summer Licensing Basis, the RADTRAD computer program following the guidance of Regulatory Guide 1.183. Dose consequences are provided for the mass transfer input from the current licensing basis and new transient analyses.

The use of the WCAP-10698-P-A methodology, longer operator action times and resulting dose consequences are being submitted for prior NRC review and approval. Upon approval, conforming changes will be made to the Virgil C. Summer Nuclear Station Final Safety Analysis Report (FSAR) and submitted to the NRC staff in accordance with 10 CFR 50.71 as part of the regular FSAR update process.

3.0 TECHNICAL EVALUATION

3.1 INTRODUCTION

3.1.1 SGTR Event Description

A steam generator tube rupture (SGTR) results in a loss of primary coolant from the reactor coolant system (RCS) to the secondary side of the affected (also called ruptured) steam generator (SG). The event is assumed to take place at full power with the reactor coolant contaminated with fission products based on a conservative assumption regarding cladding breech. The flow of radioactive reactor coolant results in contamination of the secondary system. In the event of a coincident loss of offsite power or failure of the condenser steam dump system, discharge of activity to the environment

takes place via the SG power-operated relief valves (PORVs) or safety valves. Operator actions are required to terminate the primary to secondary break flow and the release of steam to the environment. The SGTR analysis is performed to assure that the radiological consequences resulting from an SGTR are within allowable guidelines.

3.1.2 Licensing Basis SGTR Analysis

The licensing basis SGTR analysis for the Virgil C. Summer Nuclear Station (VCSNS), is presented in Chapter 15.4.3 of the Final Safety Analysis Report (FSAR). The SGTR analysis consists of simplified thermal-hydraulic calculations to determine the primary-to-secondary break flow and the steam released to the environment and a calculation of the radiological consequences resulting from the event.

The current SGTR thermal-hydraulic calculations do not include a computer analysis to determine the plant transient behavior following a SGTR. Instead, simplified calculations are performed, based on the expected SGTR transient response, to determine the primary-to-secondary break flow and the steam release to the environment for use in calculating the radiological consequences due to the event. Although the operator actions were not explicitly modeled in the analyses, it was implicitly assumed that the required operator actions to terminate the break flow and the steam release from the ruptured steam generator can be performed within 30 minutes. No single failure is utilized in the current SGTR licensing basis analysis.

3.1.3 Bases for the Proposed Change

The supplemental SGTR analyses are provided in recognition of the following:

- Improvements in the Emergency Operating Procedures (EOPs) and changes in control room protocols have resulted in longer times to complete the SGTR recovery.
- During plant simulator exercises, the operating crews are taking greater than 30 minutes to terminate primary to secondary break flow following a SGTR.
- The current SGTR methodology does not explicitly model operator actions and thus is not adequate to examine the effects of extended operator action times.
- With an extended duration of primary to secondary break flow, the potential for SG overfill increases.
- SG overfill represents a challenge to the structural integrity of the steam line, may lead to water relief out of a main steam safety/relief valve, and potentially create an un-isolable release path should a main steam safety/relief valve fail to fully reset.

As a result of the above, administrative limits on RCS activity are currently in place to ensure radiological consequences are limited should a SGTR occur pending this update to the VCSNS licensing basis to support longer operator action times.

As described in Section 3.1.4, the supplemental analyses model operator actions in accordance with the VCSNS Emergency Operating Procedures with times confirmed by plant-specific simulator exercises. The LOFTTR2 computer program, which is a modified version of Westinghouse's LOFTRAN program, is utilized. The modifications include the capability to model operator actions, an improved steam generator secondary side model, and a more realistic tube rupture break flow model.

3.1.4 Analyses to Supplement the VCSNS SGTR Analysis of Record

The supplemental SGTR analyses are performed to demonstrate margin to SG overfill and to confirm with plant-specific analyses that the simplified calculations with a 30-minute break flow and release duration produce conservatively high dose results compared to dose results calculated when realistic operator actions are taken into account. In addition, the doses with the mass transfer data based on the simplified 30-minute break flow and release duration are recalculated, to provide a consistent basis for comparison between the dose calculations performed with the supplemental analysis mass release bases.

The supplemental SGTR transient analyses were performed by Westinghouse using the LOFTTR2 computer program following the methodology developed in WCAP-10698-P-A and its Supplement 1 (References 2 and 3) with the exception that a single failure is not considered. Modifications were also made to the margin-to-overfill analysis methodology consistent with WCAP-16948-P (Reference 4) to address NSAL-07-11 (Reference 5), which identified a potential non-conservative assumption regarding the direction of conservatism for decay heat in the WCAP-10698-P-A (Reference 2) methodology for evaluating margin-to-overfill. The analyses include the simulation of the operator actions for recovery from an SGTR with reactor trip based on the VCSNS-specific EOPs (EOP-1.0 and EOP-4.0) which are based on the generic Emergency Response Guidelines.

The LOFTTR2 analyses are performed for the time period from the SGTR until the primary and secondary pressures equalized (break flow termination). In the margin-to-overfill (MTO) analysis presented in Section 3.2, the water volume in the secondary side of the ruptured SG was calculated as a function of time to demonstrate that overfill does not occur. The thermal and hydraulic analysis to develop input to the radiological consequences analysis is presented in Section 3.3. In this analysis, the primary-to-secondary break flow and the steam releases to the environment from the ruptured and intact SGs were calculated for use in determining the activity released to the environment. The mass releases were calculated with the LOFTTR2 computer code from the initiation of the event until break flow termination. From break flow termination until all releases are terminated, steam releases from the intact and ruptured SGs, determined from a mass and energy balance, were obtained from the VCSNS analysis of record. The mass transfer information was used to calculate the radiological consequences at the exclusion area boundary and low population zone and to the operators in the control room. The radiological consequences analysis is presented in Section 3.4.

3.2 MARGIN TO STEAM GENERATOR OVERFILL ANALYSIS

3.2.1 Introduction

This section includes the methods, assumptions, and inputs used to analyze the margin-to-overfill (MTO) for the SGTR event as well as the sequence of events for the recovery and the calculated results for the limiting case.

3.2.2 Input Parameters and Assumptions

The supplemental analyses support operation of VCSNS at rated thermal power with a full power average temperature (Tavg) operating range from 572.0 degrees Fahrenheit to 587.4 degrees Fahrenheit and with up to 10 percent SG tube plugging within the plant's Delta-75 Steam Generators. The SGTR flow is based on a SG tube inside diameter of 0.608 inches, and the available secondary side fill volume is 5543 cubic feet per SG.

Table 1 provides an overview of the major MTO analysis inputs in comparison to the WCAP-10698-P-A values. VCSNS specific sensitivity runs were made for the following inputs to determine the limiting values with respect to the MTO: Tavg, SG tube plugging (SGTP), Safety Injection (SI) enthalpy, Emergency Feedwater (EF) enthalpy, and decay heat. As indicated in Table 1, Safety Injection Flow was based on maximum safeguards; analysis inputs are shown in Table 2.

As described in Section 3.2.2.2, control room actions are taken to cooldown and depressurize the RCS. For cooldown, the SG PORVs on the intact SGs are credited based on a minimum as-built capacity of 803,159 lbm/hr/valve at 1000 psia. For depressurization, one of the three pressurizer PORVs is credited based on a reference design capacity of 210,000 lbm/hr of saturated steam at 2235 psig.

3.2.2.1 Design Basis Accident

The accident modeled was a double-ended break of one SG tube located at the top of the tube sheet on the outlet (cold leg) side of the SG. The location of the break on the cold side of the SG results in higher primary-to-secondary break flow than a break on the hot side of the SG. A loss of offsite power (LOOP) is assumed to occur at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

3.2.2.2 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 break flow. The operator actions for SGTR recovery are provided in the EOPs, and major actions were explicitly modeled in this analysis. The operator actions modeled include isolation of the ruptured SG (RSG),

cooldown of the Reactor Coolant System (RCS), depressurization of the RCS and termination of SI to stop primary-to-secondary break flow. These operator actions are described below.

1. Identify the ruptured SG

Based on the RCS transient response, the Operator is required to determine that a SGTR is in progress and to identify the ruptured SG. Pre-trip, the key indicators will be: decreasing RCS pressure and pressurizer level with maximum makeup, mismatches between steam and feedwater flow with increases in SG narrow range level and high steam line radiation.

2. Isolate emergency feedwater (EF) flow to a suspected ruptured steam generator

For a large SGTR, a reactor trip will occur either automatically or manually by the operator. Following reactor trip, main feedwater will be isolated and steam generator water level as indicated on the narrow range instrumentation will decrease in all steam generators. This will be followed by automatic initiation of EF. Once initiated, EF flow will begin to refill the steam generators. Since primary-to-secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the SG water level instrumentation, provides confirmation of the ruptured SG and will result in operator isolation of EF flow to the ruptured steam generator once the SG tubes are covered. Early termination of EF to the ruptured SG maximizes the SG volume available to accommodate primary-to-secondary leakage until further recovery actions are completed. The MTO analysis assumes that EF flow to the ruptured SG is isolated 6 minutes after reactor trip, by which time the ruptured SG indicated level is higher than the 26 percent narrow range span (NRS) level required for EF isolation (i.e., tubes covered) and there is a significant difference in SG level compared to the level in the intact SGs.

3. Isolate steam flow from the ruptured steam generator

Once the ruptured SG has been identified, the operators continue recovery actions by isolating steam flow from the ruptured SG. In addition to minimizing radiological releases, this also enables the operators to establish a pressure differential between the ruptured and intact SGs as a necessary step toward terminating primary-to-secondary break flow. As long as the ruptured SG main steam isolation valve (MSIV) is closed as part of RSG isolation, the time the operators close the MSIV does not impact the analysis results; so, no specific time assumption is included in the analysis.

4. Cooldown the RCS using the intact SGs

After isolation of the ruptured SG MSIV, the RCS is cooled rapidly to less than the saturation temperature corresponding to the ruptured SG pressure by dumping steam from only the intact SGs. This ensures adequate subcooling in the RCS after

depressurization to the ruptured SG pressure in subsequent actions. The margin-to-overfill analysis assumed that 15 minutes elapsed from the time of reactor trip until the cooldown was initiated. Since the analysis assumes that offsite power was lost at reactor trip, the cooldown was performed by dumping steam using the Power Operated Relief Valves (PORVs) on the intact SGs. When the EOP target temperature for cooldown (494 degrees Fahrenheit) was reached, the cooldown was terminated. The PORVs on the intact SGs were then used as necessary to maintain that temperature.

5. Depressurize the RCS to restore inventory

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

The analysis assumed that 4 minutes elapsed from the time the cooldown was terminated until the depressurization was initiated. Since offsite power was assumed to be lost at the time of reactor trip, the RCPs were not running and thus normal pressurizer spray was not available. Therefore, the depressurization was modeled using a pressurizer PORV.

The RCS depressurization is continued until any of the following conditions in the VCSNS EOP for SGTR are satisfied: RCS pressure is less than the ruptured SG pressure and pressurizer level is greater than 10 percent, or pressurizer level is greater than 76 percent, or RCS subcooling is less than 52.5 degrees Fahrenheit.

6. Terminate SI to stop primary-to-secondary break flow

The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient RCS inventory to ensure that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary-to-secondary break flow. The analysis assumed that 4 minutes elapsed from the time the depressurization was terminated until SI could be stopped. SI can be stopped provided the following conditions in the VCSNS EOP for SGTR are satisfied: RCS pressure is stable or rising, pressurizer level is greater than 10 percent, RCS subcooling is greater than 52.5 degrees Fahrenheit, and a secondary heat sink is confirmed.

Consistent with the Reference 2 methodology, the analysis does not model EOP actions after SI termination leading to break flow termination. The primary-to-secondary break flow continues after the SI flow is stopped until the RCS and ruptured SG pressures equalize. This provides a conservative margin-to-overfill result.

The assumed operator action times (to isolate EF flow to the ruptured SG, to initiate RCS cooldown, to initiate RCS depressurization, and to terminate SI) bound actual crew performance based on simulator exercises. Table 3 summarizes the operator action times assumed in the analysis.

3.2.3 Description of Analysis

The LOFTTR2 analysis for the limiting margin-to-overfill case is described below. The limiting case with respect to margin to SG overfill considered operation at the maximum operating temperature (587.4 degrees Fahrenheit), with the minimum main feedwater temperature (435.0 degrees Fahrenheit), and the maximum SGTP level (10 percent). The sequence of events for this transient is presented in Table 4.

Following the tube rupture, water flowed from the primary into the secondary side of the ruptured SG since the primary pressure was greater than the SG pressure. In response to this loss of coolant the RCS pressure decreased as shown in Figure 1, as the steam bubble in the pressurizer expanded. As the RCS pressure decreased due to the continued primary-to-secondary break flow, automatic reactor trip occurred on an Overtemperature-delta T (OTDT) trip signal. After reactor trip, core power rapidly decreased to decay heat levels. The turbine stop valves closed and steam flow to the turbine was terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remained 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 caused the secondary side pressure to increase rapidly after reactor trip, as shown in Figure 1, until the SG PORVs (and safety valves if their setpoints are reached) lifted to dissipate the energy. As a result of the assumed loss of offsite power, main feedwater flow was assumed to be terminated and EF flow was assumed to be automatically initiated following reactor trip.

The RCS pressure and pressurizer level would continue to decrease after reactor trip as energy transfer to the secondary system shrinks the primary coolant and the tube rupture break flow continues to deplete primary inventory. The operators initiated an SI signal in anticipation that it would be required by the decrease in RCS inventory. The SI flow increased the RCS inventory and stabilized the RCS pressure. The RCS pressure then trended toward the equilibrium value where the SI flow rate would equal the break flow rate.

EF flow to the ruptured SG was isolated 6 minutes after the reactor trip. By this time the ruptured SG level was well above the level required for isolation, and well above the level in the intact SGs, as shown in Figure 3. The operators then proceed to isolate steam flow from the ruptured SG, completing isolation prior to initiating the cooldown.

The PORVs on the intact SGs were opened for the RCS cooldown at 15 minutes after the reactor trip. (The analysis models closure of the ruptured SG MSIV immediately before the cooldown was initiated.) The cooldown was continued until the EOP cooldown termination temperature (494 degrees Fahrenheit) was reached. When this condition was satisfied, the operator closed the SG PORVs to terminate the cooldown. This cooldown ensured that there would be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured SG pressure. The reduction in the intact SG pressure required to accomplish the cooldown is shown in Figure 1. The RCS pressure also decreased during this cooldown process due to shrinkage of the RCS as shown in Figure 1.

After termination of the cooldown, a 4-minute operator action time was imposed prior to the RCS depressurization. In this analysis, the RCS depressurization was terminated when the RCS pressure was reduced to less than the ruptured SG pressure and the pressurizer level was above the required value, since there was adequate subcooling margin and the high pressurizer level setpoint was not reached. The RCS depressurization is shown in Figure 1. The depressurization reduced the break flow as shown in Figure 2.

After termination of the depressurization, a 4-minute operator action time was imposed prior to SI termination. The SI flow was terminated at this time since the requirements for SI termination were satisfied (RCS subcooling was greater than the required allowance for subcooling uncertainty, minimum EF flow was available or at least one intact SG level was in the narrow range, the RCS pressure was stable or increasing, and the pressurizer level was greater than the required value).

After SI termination the RCS pressure began to decrease as shown in Figure 1. Break flow continues for more than 16 minutes after SI flow is stopped, as shown in Table 4 and Figure 2, further reducing the margin-to-overfill. Break flow is quickly terminated once the PORVs on the intact steam generators are re-opened to maintain the core exit temperature.

3.2.4 Results

The primary-to-secondary break flow rate throughout the recovery operations is presented in Figure 2. The water volume in the ruptured SG is presented as a function of time in Figure 4. The peak ruptured SG water volume is 5532 cubic feet resulting in 11 cubic feet of margin-to-overfill. Therefore, it is concluded that overfill of the ruptured SG will not occur for a design basis SGTR.

It is recognized that there are a number of conservative inputs that could be relaxed to increase the margin-to-overfill. These include reduced operator action times, improved and more detailed modeling of operator actions to reduce/control RCS pressure after SI termination, less conservative EF flow splits, realistic EF initiation delays, and crediting a portion of the steamline to show greater margin to water release.

3.2.5 Conclusions

It is concluded that overfill of the ruptured SG will not occur for a design basis SGTR for VCSNS.

Table 1 Margin-to-Overfill Inputs

Parameter	WCAP-10698 Modeling Direction of Conservatism	VCSNS SGTR MTO Analysis
Initial Conditions		
Power	Full - Power [Nominal + Uncertainty]	Full - Power [Nominal + Uncertainty](1)(2)
RCS Pressure	Minimum	Minimum
Pressurizer Water Level	Maximum	Maximum
SG Secondary Mass	Maximum	Maximum(3)
Break Location	Cold Leg	Cold Leg
Offsite Power Availability		
Offsite Power	Loss of Offsite Power (LOOP)	LOOP(4)
Protection Setpoints and Errors		
Reactor Trip Delay	Minimum	Minimum
Turbine Trip Delay	Minimum	Minimum
SG Relief Setpoint	Minimum (PORV)	Minimum (PORV)
Pressurizer Pressure Trip Setpoint	Low Pressurizer Pressure	OTΔT
Pressurizer Pressure SI Setpoint	Maximum	Maximum(5)
Safeguards Capacity		
SI Flow Rate	Maximum	Maximum(5)
SI Delay	Minimum	Minimum(5)
SI Temperature	-	Maximum(6)
EF Flow Rate (Isolation by Operator)	Maximum	Maximum
EF System Delay	Minimum	Minimum(7)
EF Temperature	Maximum	Minimum(8)
Control Systems		
CVCS Operation, PZR Control	Not Operating	Not Operating
Turbine Runback Mass Penalty	Included	Included
Decay Heat		
Decay Heat	1971 Nominal	1979-2σ(9)
Single Failure		
Single Failure	Included	Not Included(10)

Table 1 (continued)

Notes:

1) Full power Tavg was assumed to range from 572.0 to 587.4 degrees Fahrenheit. Plant specific sensitivity studies show it is more conservative (i.e., less margin-to-overfill) to use the high value.
2) SG tube plugging was assumed to range from 0 to 10 percent. Plant specific sensitivity studies show it is more conservative (i.e., less margin-to-overfill) to use the high value.
3) The plant was assumed to be operating with the feedwater temperature at the low end of the temperature range (435 degrees Fahrenheit) since this results in a higher mass of water in the SG at the start of the event, which limits the amount of break flow and emergency feedwater (EF) that can accumulate in the ruptured SG without forcing water into the steam lines.
4) A Loss of Offsite Power (LOOP) is assumed to occur at reactor trip resulting in a coastdown of the Reactor Coolant Pumps.
5) High Head Safety Injection (HHSI) is assumed to be initiated coincident with reactor trip and loss of offsite power with zero delay. The maximum safeguards flow shown in Table 2 was utilized.
6) HHSI temperature is assumed to range from 40 to 120 degrees Fahrenheit. Plant specific sensitivity studies show it is more conservative (i.e., less margin-to-overfill) to use the high value.
7) Emergency Feedwater (EF) is assumed to be initiated coincident with reactor trip and loss of offsite power with zero delay. All three pumps are assumed to operate, and the flow split was selected to maximize flow to the ruptured SG. No purge volume was credited to delay delivery of cold EF to the SGs as this minimizes steam release. Flow to the ruptured SG continues until it is isolated by the operator at the defined time after reactor trip, whereas flow to the intact SGs was assumed to be throttled to maintain level at or below 60 percent of the narrow range span.
8) EF temperature is assumed to range from 40 to 120 degrees Fahrenheit. Plant specific sensitivity studies show it is more conservative to use the low value.
9) Plant specific sensitivity studies show it is more conservative to use a low value of decay heat in the MTO analysis. This finding is consistent with NSAL-07-11 (Reference 5).
10) Although a deviation from WCAP-10698-P-A (Reference 2), neglecting the effects of a single failure is consistent with the VCSNS licensing basis for SGTR.

Table 2 Safety Injection Flows for Design Basis SGTR Analysis (Maximum Safeguards)	
Pressure (psig)	**Total Injection Flow Rate (lbm/sec)**
1100	93.6
1200	91.0
1300	88.2
1400	85.3
1500	82.4
1600	79.4
1700	76.2
1800	72.9
1900	69.5
2000	65.8
2100	62.0
2200	57.9
2300	53.6
2400	48.8
2500	43.6
2600	37.6
2700	30.6
2800	21.6

Table 3
Operator Action Times For Design Basis SGTR Analysis[1]

Action	Time
Operator action time to isolate EF flow to ruptured SG	Margin-to-Overfill: 6 minutes from reactor trip or when level reached 26% NRS, whichever is later Input to Dose: Time for the level to reach 26% NRS[2]
Operator action to isolate MSIV on ruptured SG[3]	Margin-to-Overfill: just prior to cooldown initiation Input to Dose: immediately after reactor trip
Operator action time to initiate cooldown	Margin-to-Overfill: 15 minutes from reactor trip Input to Dose: 30 minutes from reactor trip[4]
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 minutes from end of cooldown
Depressurization	Calculated by LOFTTR2
Operator action time to terminate SI following depressurization	Maximum of 4 minutes from end of depressurization or time to satisfy termination criteria
Pressure equalization	Calculated by LOFTTR2

1. The operator action times modeled in the analysis have been validated by VCSNS simulator exercises (see Attachment 1) and are used as acceptance criteria for continuing training.

2. The analysis conservatively allows EF isolation to occur immediately after initiation.

3. This does not impose a requirement on the operators.

4. The time used in the input to dose analysis was extended to provide a more conservative result. This extends the duration of break flow, break flow flashing and ruptured SG releases which increases the calculated doses and strengthens the conclusion that the doses based on the simplistic calculation with 30 minute break flow and release duration are conservative.

Table 4
Sequence of Events for Limiting Margin-to-Overfill Analysis

Event	Time (seconds)
Steam Generator Tube Rupture	0
Reactor Trip (OTDT) and LOOP	84
EF Initiated	85
SI Actuated (Manually)	85
EF Flow to Ruptured SG Isolated	445
Ruptured SG MSIV Closed	986
RCS Cooldown Initiated	988
RCS Cooldown Terminated	1442
RCS Depressurization Initiated	1684
RCS Depressurization Terminated	1752
SI Terminated	1993
Break Flow Terminated	3006

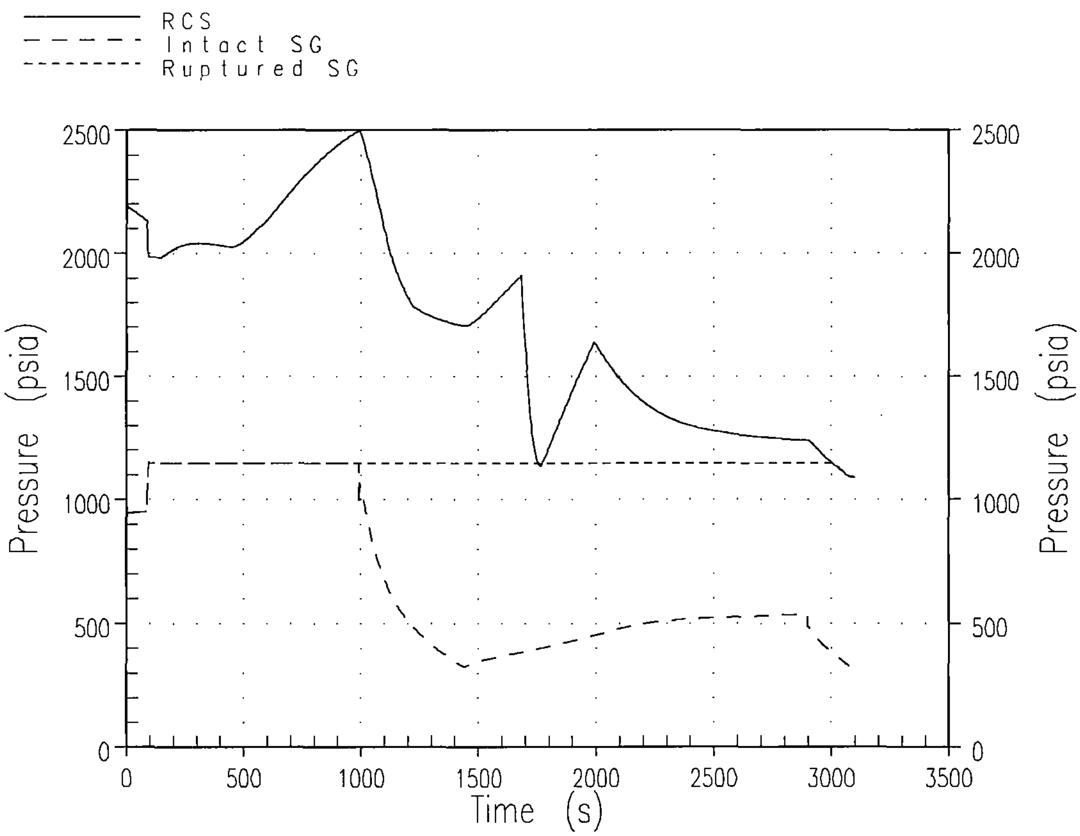

Figure 1
RCS and Secondary Pressures –
Margin-to-Overfill Analysis

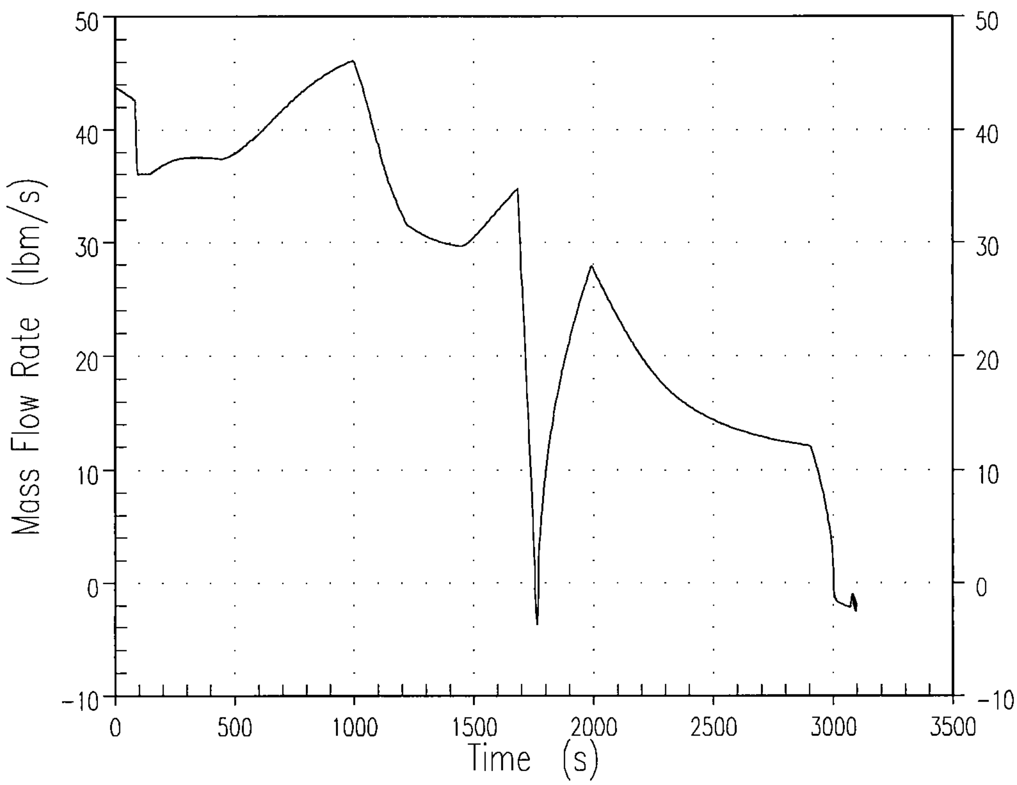

Figure 2
Primary to Secondary Break Flow –
Margin-to-Overfill Analysis

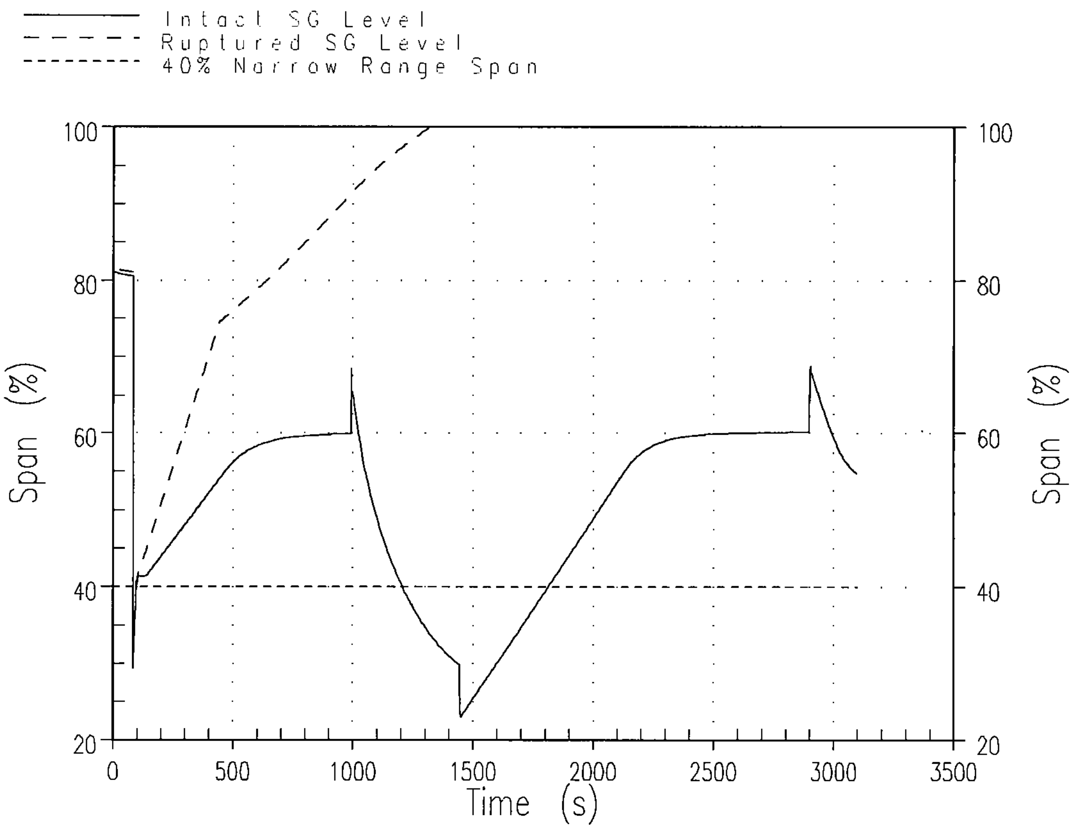

Figure 3
Indicated SG Levels –
Margin-to-Overfill Analysis

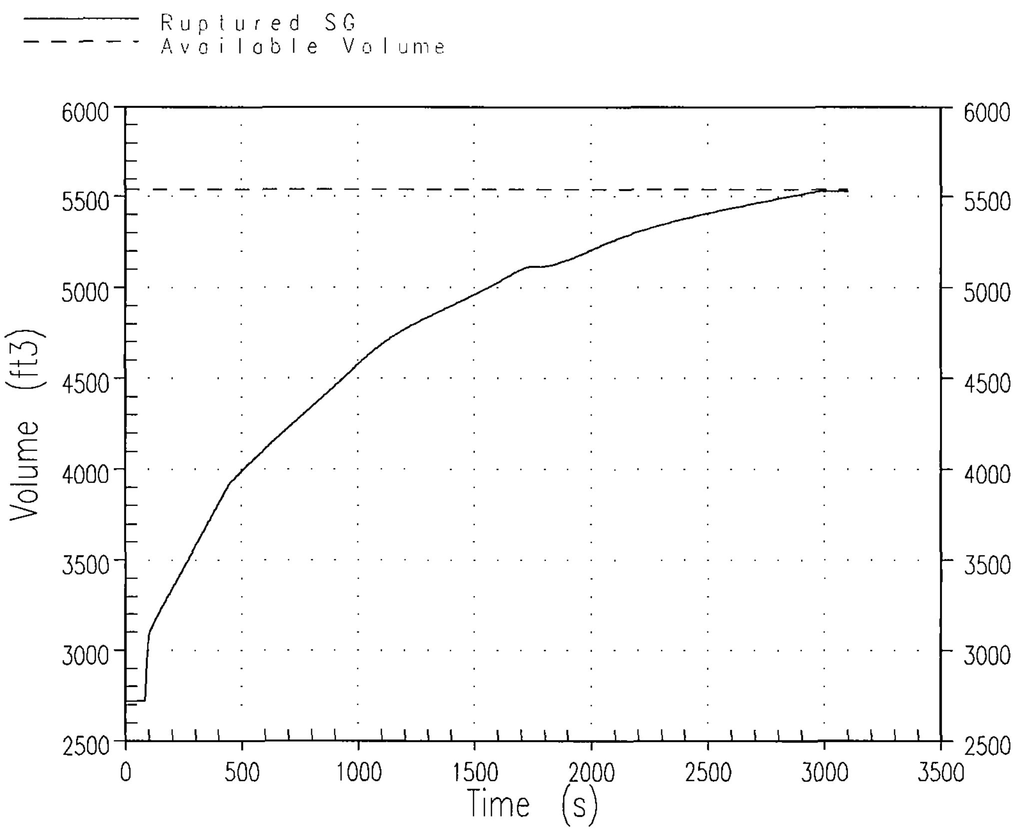

Figure 4
Ruptured SG Water Volume –
Margin-to-Overfill Analysis

3.3 THERMAL AND HYDRAULIC ANALYSIS FOR RADIOLOGICAL CONSEQUENCES

3.3.1 Introduction

An additional thermal and hydraulic analysis was performed to determine the input for the radiological consequences analysis for an SGTR event modeling the operator responses and break flow continuing beyond the 30 minutes considered in the licensing basis analysis. The thermal and hydraulic analysis was performed by Westinghouse using the LOFTTR2 program and the methodology developed in References 2 and 3, with the exception that a single failure was not considered. This exception is consistent with the VCSNS licensing basis. Plant-specific parameters and operator actions consistent with the VCSNS EOPs and operator training were used. This section includes the methods, assumptions and input used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

3.3.2 Input Parameters and Assumptions

Plant specific sensitivity runs were made to determine the limiting Tavg and Steam Generator Tube Plugging assumptions with respect to the thermal-hydraulic analysis to determine the mass releases for the radiological consequences analysis. The limiting analysis modeled the plant operating at the high end of the Tavg range (587.4 degrees Fahrenheit) with no Steam Generator Tube Plugging. Plant specific sensitivity studies show that a higher initial secondary water mass resulted in increased steam releases and flashed break flow. Assuming feedwater temperature at the high end of the temperature range (445 degrees Fahrenheit), the initial SG mass was increased to account for uncertainty.

Other plant inputs and assumptions are the same as those described in Section 3.2.2 and Table 1 for the MTO analysis except for the following:

- Consistent with WCAP-10698-P-A, decay heat was maximized by using the 1971+20 percent ANS decay heat model.
- Consistent with WCAP-10698-P-A, EF temperature was assumed to be at a maximum (i.e., 120 degrees Fahrenheit).
- Minimum EF flow was delivered to the SGs following reactor trip and loss of offsite power with a maximum 60-second delay. The flow split was selected to minimize flow to the ruptured SG. A maximum purge volume of 68 cubic feet was modeled to delay delivery of cold EF to the SGs and maximize steam release.

3.3.2.1 Design Basis Accident

The accident modeled was a double-ended break of one SG tube located at the top of the tube sheet on the outlet (cold-leg) side of the SG. The location of the break results in higher primary to secondary break flow than a break on the hot side of the SG, as determined by Reference 2. However, the break flow flashing fraction was

conservatively calculated assuming all of the break flow comes from the hot-leg side of the SG consistent with Reference 3.

3.3.2.2 Operator Action Times

The major operator actions required for the recovery from an SGTR are discussed in Section 3.2.2.2, and the operator action times used for the analysis are presented in Table 3. The operator action times assumed for the MTO analysis were also used for the radiological consequences analysis with the following exceptions:

1. Isolate EF flow to the ruptured SG

Earlier EF isolation results in higher releases. Consequently, it was assumed that EF flow to the ruptured SG was isolated when level in the SG reached the required level that assures the SG tubes are covered. The analysis conservatively allows EF isolation to occur immediately after initiation if the level in the ruptured SG is sufficiently high.

2. Isolate steam flow from the ruptured SG

Once the ruptured SG has been identified, operators continue recovery actions by isolating steam flow from the ruptured SG. The input to dose analysis assumes the MSIV on the ruptured SG is closed immediately after reactor trip so that any steam flow out of the ruptured SG is to the environment.

3. Cooldown the RCS using the intact SGs

The input to dose analysis assumed that 30 minutes elapsed from the time of reactor trip until the cooldown was initiated. This additional conservatism strengthens the Section 3.4 conclusion that the licensing basis analysis provides a conservative prediction for the radiological consequences of a SGTR.

3.3.3 Description of Analysis

The LOFTTR2 results for the limiting input to dose analysis are presented below. As described above, the limiting case is based on plant operation at the maximum operating temperature (587.4 degrees Fahrenheit), with the maximum main feedwater temperature (445.0 degrees Fahrenheit) and the minimum Steam Generator Tube Plugging level (0 percent).

The sequence of events for this transient is presented in Table 5. Although the times change, the general evolution of the event is as described in Section 3.2.3. Figures 5 through 10 show transient results for primary and secondary pressure, primary to secondary break flow, steam release rate to the environment, break flow flashing fraction, total flashed break flow and ruptured SG water volume.

3.3.4 Results

The primary-to-secondary break flow rate throughout the recovery operations is presented in Figure 6. The break flow flashing fraction was calculated using the primary pressure, the ruptured loop hot leg temperature, and the ruptured SG secondary pressure. The flashing fraction is presented in Figure 8, and the total flashed break flow is presented in Figure 9. The ruptured SG PORV steam release rate is presented in Figure 7, along with the total intact SG PORV steam release rate. The ruptured steam water volume is shown in Figure 10. The water volume in the ruptured SG when the break flow is terminated is significantly less than the available SG volume.

For the time period from initiation of the accident until break flow termination, the releases were determined from the LOFTTR2 results. Since the condenser was in service until reactor trip, any radioactivity released to the environment prior to reactor trip would be through the condenser vacuum exhaust. A conservatively high pre-trip steam flow rate is considered for use in the dose analysis. After reactor trip, the releases to the environment were assumed to be from the SG PORVs.

The transfer and release data are presented in Table 6 and Table 7. The mass transfer data tabulated for use in the dose analysis conservatively added 10 percent to the mass transfer data calculated by LOFTTR2 and presented in the figures. This added conservatism strengthens the Section 3.4 conclusion that the licensing basis analysis provides a conservative prediction for the radiological consequences of a SGTR.

Low values for the RCS and SG mass data that bound the LOFTTR2 results were developed for use in the dose analysis. The data is presented in Table 8. Although the mass transfer data presented was obtained from a transient modeling maximum initial secondary side water mass, a low SG mass was selected to bound scenarios modeling the minimum initial secondary side water mass.

3.3.5 Conclusions

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis were completed and the data was tabulated for the limiting cases. The results of the analysis are used as input to the radiological consequences analysis presented in Section 3.4.

Table 5
Sequence of Events for Limiting Input to Radiological Consequences Analysis

Event	Time (seconds)
Steam Generator Tube Rupture	0
Reactor Trip (OTDT) and LOOP	76
SI Actuated (Manually)	76
Ruptured SG MSIV Closed	77
EF Initiated	136
EF Flow to Ruptured SG Isolated	137
RCS Cooldown Initiated	1878
Break Flow Stops Flashing	2067
RCS Cooldown Terminated	2429
RCS Depressurization Initiated	2670
RCS Depressurization Terminated	2732
SI Terminated	2972
Break Flow Terminated	3411

Table 6 Break Flow and Flashed Break Flow			
Start of Period (sec)	**End of Period (sec)**	**Integrated Break Flow during Period (lbm)**	**Integrated Flashed Break Flow during Period (lbm)**
0	76	3600	610
76	2067	94240	5120
2067	3411	37880	0
3411	7200	0	0
7200	28800	0	0
28800	86400	0	0

Table 7
Intact and Ruptured SG Steam Flow to Environment

Start of Period (sec)	End of Period (sec)	Integrated Intact SGs Steam Flow to Environment during Period (lbm)	Integrated Ruptured SG Steam Flow to Environment during Period (lbm)
0	76	199120*	99560*
76	3411	348300	73740
3411	7200	256400**	0
7200	28800	924900**	0
28800	86400	1200000**	0

* Pre-trip steam releases are through the condenser. The values listed here are based on a bounding steam flow rate of 1310 lbm/sec/SG.

** The LOFTTR2 analysis was performed to provide transient mass transfer data until break flow termination. The operators will then continue the SGTR recovery actions and the plant is cooled and depressurized to cold shutdown conditions. Intact SG releases after break flow termination were obtained from the licensing basis analysis.

Table 8
RCS and SG Mass Data

	Liquid Mass* (lbm)
Initial RCS	3.4E+5
Initial Ruptured SG	9.6E+4
Initial Intact SGs (total)	1.9E+5

* These are conservative minimum values. The RCS mass bounds that calculated for the range of Tavg considered. The SG masses are the nominal initial secondary water masses reduced to address uncertainty, even though the transient calculations were performed using the nominal initial secondary water masses increased to address uncertainty as noted in Section 3.3.4.

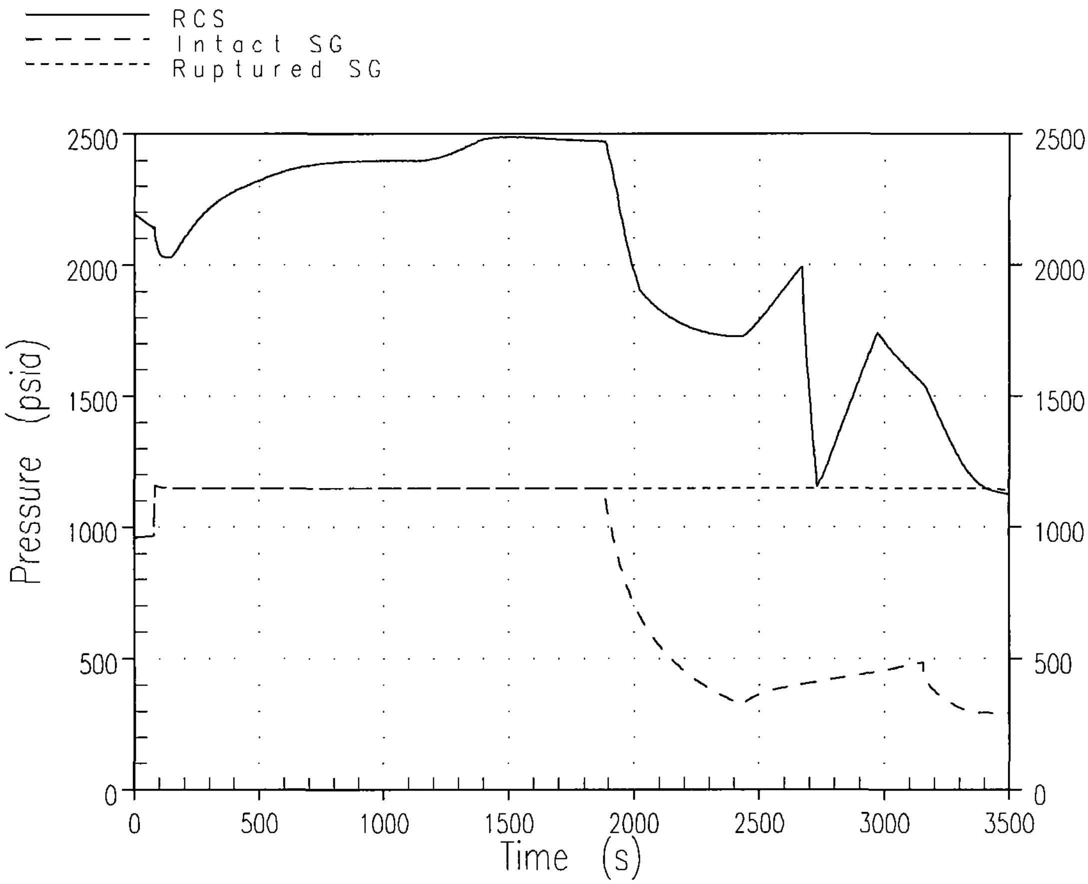

Figure 5
RCS and Secondary Pressure –
Input to Radiological Consequences Analysis

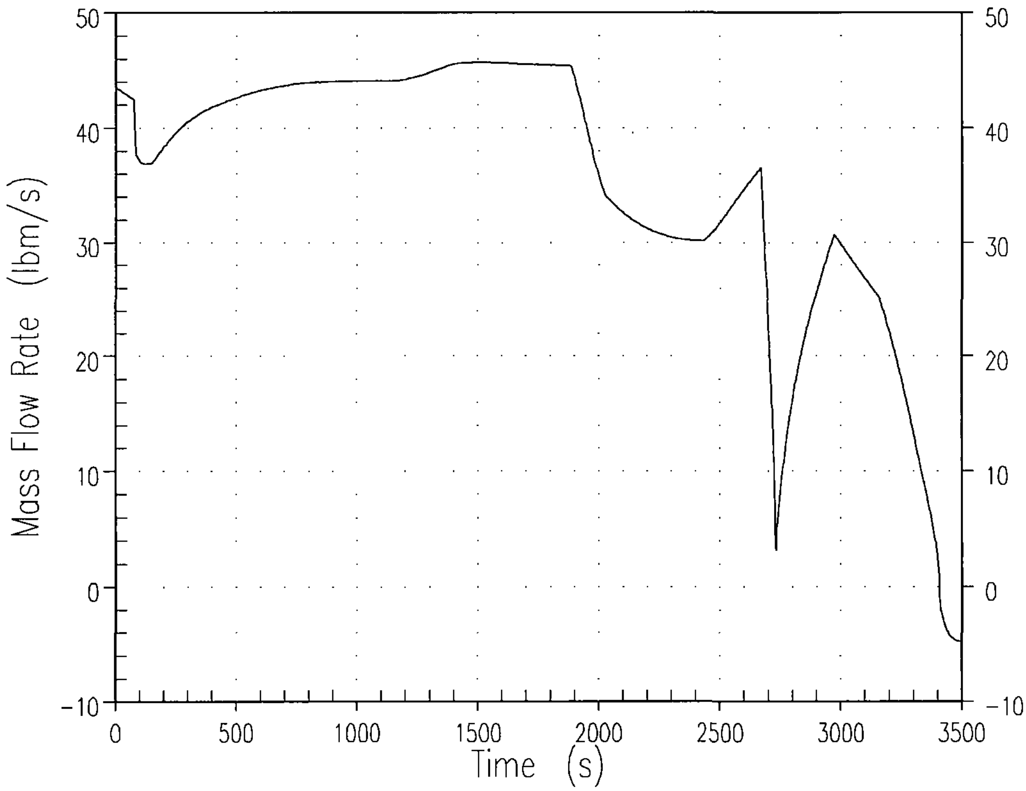

Figure 6
Primary to Secondary Break Flow –
Input to Radiological Consequences Analysis

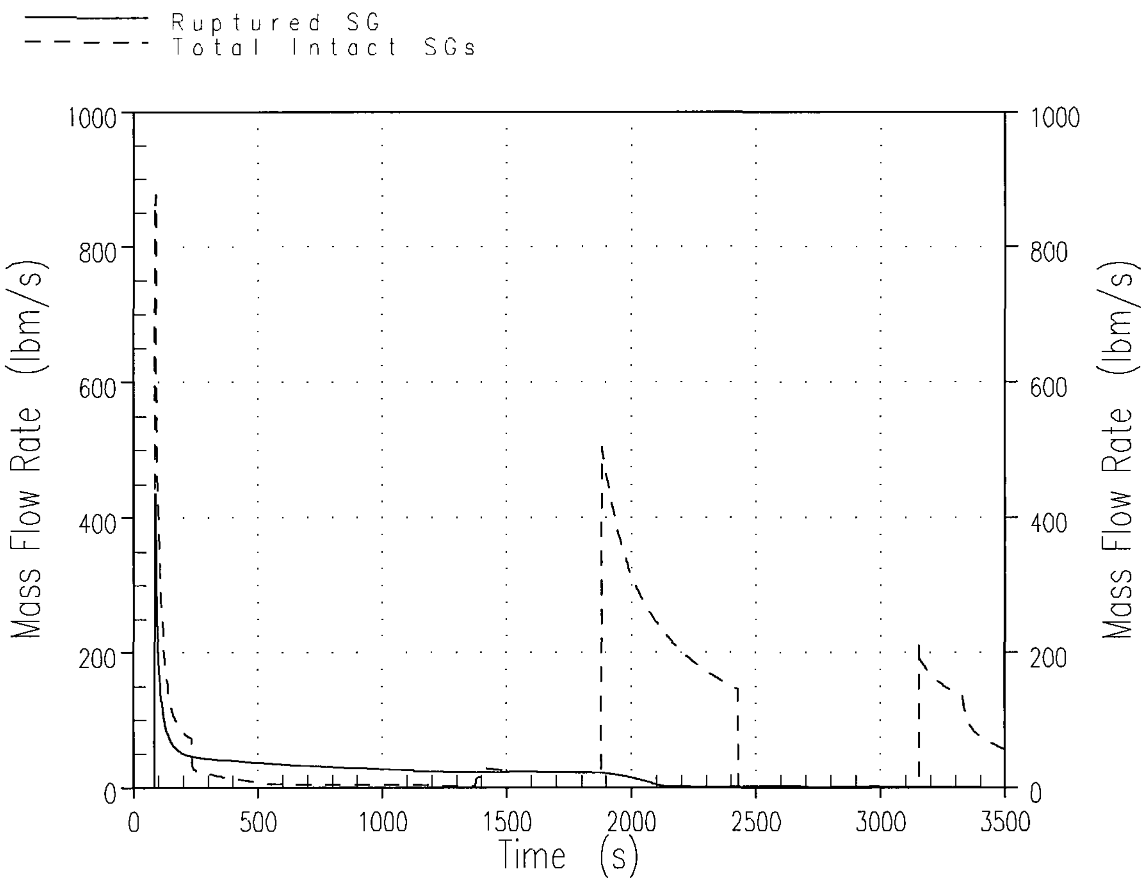

Figure 7
SG Steam Release Rate to the Environment –
Input to Radiological Consequences Analysis

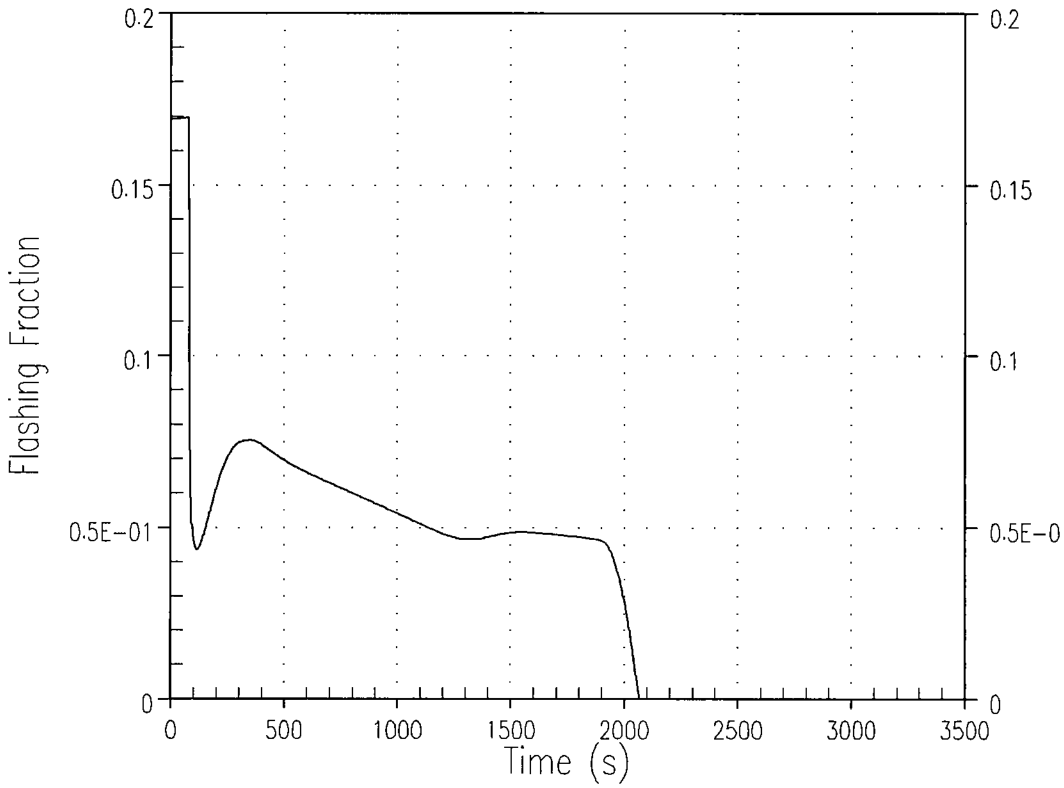

Figure 8
Break Flow Flashing Fraction –
Input to Radiological Consequences Analysis

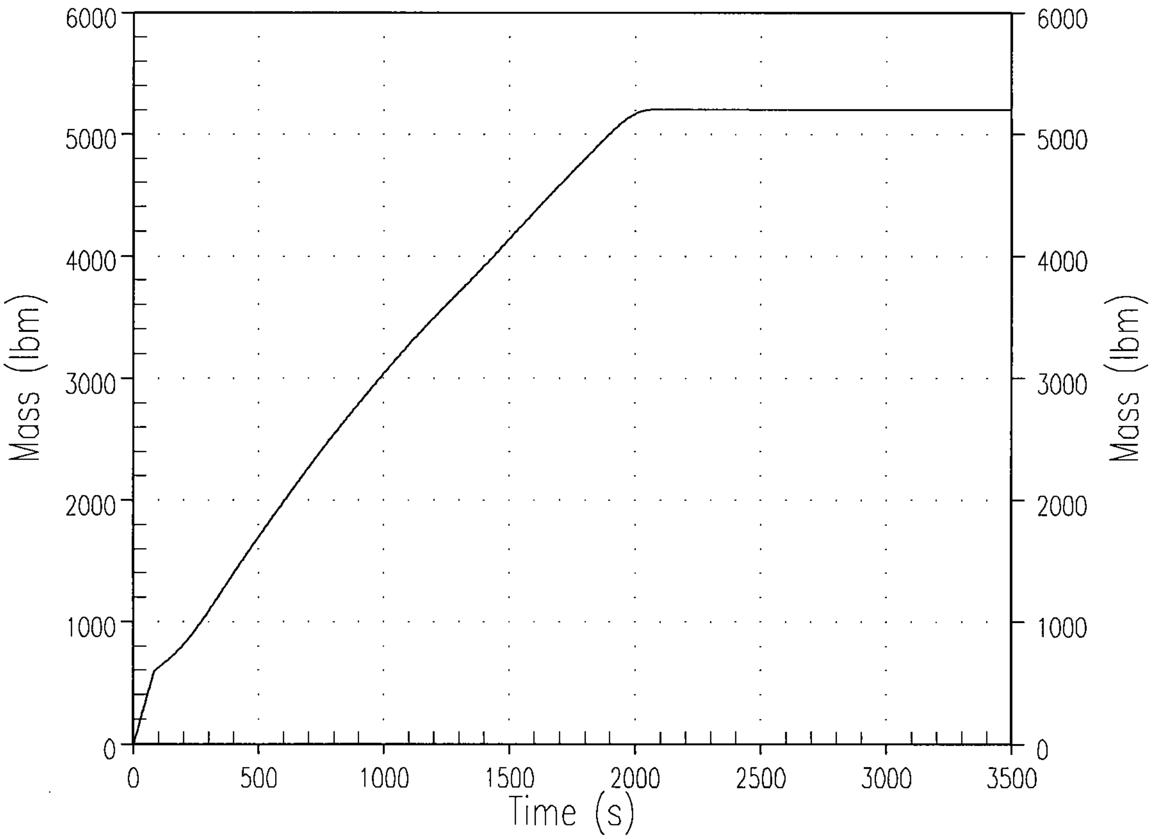

Figure 9
Total Flashed Break Flow –
Input to Radiological Consequences Analysis

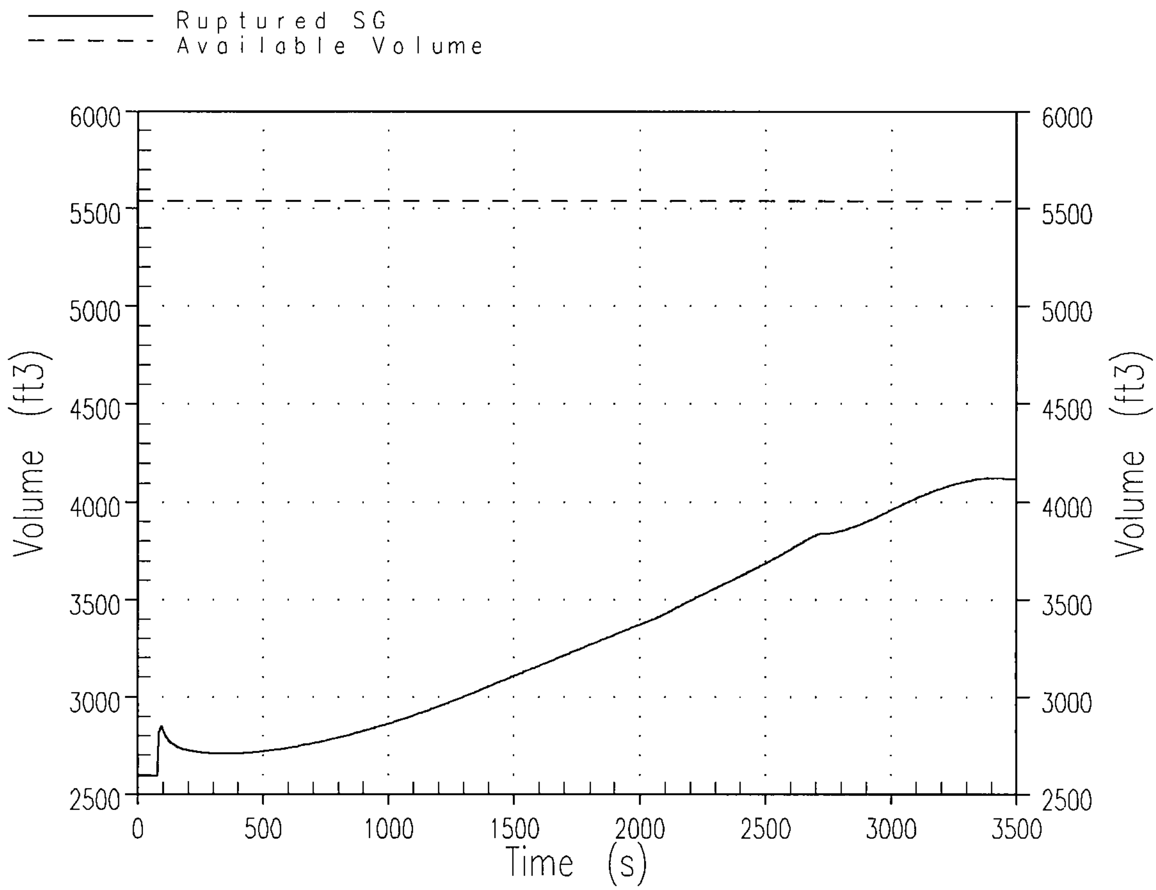

Figure 10
Ruptured SG Water Volumes –
Input to Radiological Consequences Analysis

3.4 RADIOLOGICAL CONSEQUENCES

3.4.1 Introduction

The evaluation of the radiological consequences of an SGTR assumes that the reactor has been operating at the Technical Specification limits for primary coolant activity and primary-to-secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the secondary coolant. During an SGTR, activity from the primary coolant enters the secondary coolant via the ruptured tube and pre-existing primary-to-secondary leakage and is released to the environment through the SG safety valves, SG PORVs and the condenser air removal system exhaust.

The amount of radioactivity released to the environment due to an SGTR depends upon primary and secondary coolant activities, iodine spiking effects, primary-to-secondary break flow, break flow flashing, 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 SG and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively modeled for a design basis double-ended rupture of a single SG tube.

Section 3.4 presents the calculation of offsite and control room doses for two sets of SGTR mass release data. One set of mass transfer data is consistent with that used in the licensing basis SGTR dose analysis approved in Reference 1, based on the simplified 30-minute break flow and release duration. The second set of mass transfer data is that presented in Section 3.3 of this report. A comparison of the calculated doses is made to demonstrate that the mass release data used in the licensing basis SGTR dose analysis is conservative even though it does not consider break flow continuing beyond 30 minutes.

The following sections discuss the methods and assumptions used to analyze the radiological consequences of the SGTR event, as well as the calculated results.

3.4.2 Input Parameters and Assumptions

Major assumptions and parameters are discussed in this section and summarized in Table 9. Consistent with VCSNS current licensing basis (Reference 1), the SGTR radiological consequences analysis was performed following the guidance provided in Regulatory Guide (RG) 1.183 (Reference 8) and using the RADTRAD code, Version 3.03 (Reference 9). The calculations determined the doses based on a pre-accident iodine spike and on an accident-initiated iodine spike.

3.4.2.1 Mass Transfer Assumptions

Two data sets (i.e., break flow, flashed break flow, and steam releases from the intact and ruptured SGs) were prepared to support dose calculations for the licensing basis

analysis and the thermal and hydraulic analysis in Section 3.3. The RCS is modeled with a constant mass at the initial value listed in Table 8 in both sets of calculations. The calculations for the licensing basis analysis model the SG secondary with constant masses at a nominal value consistent with the basis of the mass transfer data. The calculations for the Section 3.0 analysis model the SG secondary with constant masses at the initial values listed in Table 8.

The total primary-to-secondary leak rate to the intact SGs was assumed to be 1 gallon per minute. The leakage to the intact SGs was assumed to persist for the duration of the accident. Cold conditions were assumed in selecting the density of 62.4 lbm/ft^3 to model the mass flow rate (8.34 lbm/min) for this leakage.

3.4.2.2 Source Term Assumptions

The radionuclide concentrations in the primary coolant and secondary coolant prior to and following the SGTR, were based on the following:

1. There is no fuel damage as a result of the postulated SGTR. The iodine concentrations in the RCS were based upon either a pre-accident iodine spike or an accident-initiated iodine spike as outlined in RG 1.183 (Reference 8).

    a. Pre-Accident Spike - A reactor transient occurred prior to the SGTR and raised the primary coolant iodine concentration to 60 µCi/gm Dose Equivalent (DE) I-131 which is the Technical Specification limit for iodine concentration excursions beyond equilibrium conditions.

    b. Accident-Initiated Spike - The primary coolant iodine concentration was initially at the equilibrium operation Technical Specification limit of 1.0 µCi/gm DE I-131. Coincident with the SGTR, an iodine spike was initiated. This spike increased the iodine release rate from the fuel to the coolant to a value 335 times greater than the release rate corresponding to the initial RCS iodine concentration. The spike was assumed to continue until 8 hours after the start of the event.

2. The initial secondary coolant iodine concentration was 0.1 µCi/gm DE I-131. This is the Technical Specification limit.

3. The initial concentration of noble gases, alkali metals, and bromine in the RCS was based on operation with 1 percent of the average fuel rods containing cladding defects.

4. No noble gases were present in the secondary coolant at the start of the event since retention of noble gases in water is negligible.

5. The initial concentration of alkali metals and bromine in the secondary coolant was 10 percent of the concentration initially in the RCS, reflecting the ratio of the Technical Specifications limits on DE I-131.

6. The chemical form of iodine released from the SGs to the environment was 97percent elemental and 3 percent organic consistent with the guidance of RG 1.183 (Reference 8).

The iodine, bromine, alkali metal and noble gas concentration data and the equilibrium iodine appearance rates are presented in Table 10.

3.4.2.3 Additional Assumptions for Dose Calculations

The iodine transport model used in this analysis accounted for break flow flashing, steaming, and partitioning. The model assumed that a fraction of the iodine carried by the break flow became airborne immediately due to flashing. All of the iodine in the flashed break flow was assumed to be transferred instantly out of the SG. The fraction of iodine in the break flow that was not assumed to become airborne immediately mixed with the secondary coolant and was assumed to become airborne at a rate proportional to the steaming rate. The water/steam iodine partition coefficient of 100 from RG 1.183 (Reference 8) was used. Although prior to reactor trip all releases are through the condenser, partitioning and retention in the condenser was conservatively ignored. This model is applied to bromine and alkali metals as well.

All noble gases in the break flow and primary-to-secondary leakage were assumed to be transferred instantly out of the SG to the environment.

Decay of radioactivity was credited in the fuel, the RCS and the SGs prior to release. No credit was taken for the radioactive decay during release and transport or for cloud depletion by ground deposition during transport after release to the environment. Decay of activity in the control room (CR) was credited.

Atmospheric dispersion factors (χ/Q) for the exclusion area boundary (EAB), the low population zone (LPZ), and the CR that were used to model the spread of the released activity from the release point to the receptor are presented in Table 11.

Consistent with the guidance of RG 1.183 (Reference 8) the breathing rate of 3.5×10^{-4} m^3/sec was applied for the EAB and CR for the duration of the calculation, while the LPZ dose credited a reduction to 1.8×10^{-4} m^3/sec after 8 hours until 24 hours, at which time the releases were terminated.

Total effective dose equivalent (TEDE) doses were calculated at the EAB, the LPZ and in the CR. TEDE is the sum of the committed effective dose equivalent (CEDE) dose from inhalation and effective dose equivalent (EDE) dose from external exposure. The inhalation dose conversion factors (DCFs) from Reference 10 and external exposure DCFs from Reference 11 were used in determining the dose resulting from the released activity. Values used are presented in Table 12.

The EAB doses were determined for the limiting 2-hour time interval which is the first 2 hours following the SGTR since break flow was terminated within the first 2 hours. The LPZ doses were determined for the 24-hour duration of releases.

The CR ventilation system was assumed to be placed in the emergency mode of operation 30 minutes after the initiation of the SGTR. Additional sensitivity calculations were performed crediting the emergency mode of operation 60 seconds after the SI signal. Although all releases were terminated at 24 hours when the residual heat removal (RHR) system was put in service, the calculation of CR doses was continued for 30 days to account for additional doses due to continued occupancy of the CR. The inflow (filtered and unfiltered) to the CR and the CR filtered recirculation flow, were used in the calculation of the CR doses. CR parameters used in the analysis are presented in Table 13.

3.4.3 Results

The doses for both sets of mass transfer data are presented in Table 14 compared to the limits. All limits are met with significant margin. The doses calculated for the licensing basis mass transfer data, for both spike scenarios at all locations, are at least 1.5 times higher than the doses calculated with the transient data calculated in Section 3.3. This confirms that the simplified calculations with a 30-minute break flow and release duration used in the licensing basis analysis produce conservatively high dose results. This is the case even though the transient calculations included an extra 15-minute delay prior to operator initiation of the cooldown compared to the expected operator response time credited in the margin-to-overfill analysis and validated on the plant simulator. The inclusion of the 10 percent mass transfer conservatism noted in Section 3.3.4 further strengthens this conclusion.

The limiting offsite TEDE doses for the pre-accident iodine spike, calculated using the licensing basis mass transfer data, are 0.68 rem at the EAB and 0.28 rem at the LPZ compared to the limit of 25 rem TEDE. The limiting offsite TEDE doses for the accident-initiated iodine spike, calculated using the licensing basis mass transfer data, are 0.31 rem at the EAB and 0.14 rem at the LPZ compared to the limit of 2.5 rem TEDE. The limiting control room TEDE dose, calculated using the licensing basis mass transfer data, is 1.3 rem, compared to the limit of 5.0 rem TEDE. This was calculated for the pre-accident iodine spike, with control room isolation at 30 minutes.

The sensitivity to CR isolation timing quantifies the conservatism built into the licensing basis analysis by delaying the time that the CR ventilation system was assumed to be placed in the emergency mode of operation until 30 minutes after the initiation of the SGTR. The CR doses are at least 3.5 times higher with this assumption compared to the doses calculated crediting the emergency mode of operation 60 seconds after the SI signal.

3.4.4 Conclusions

The doses at the EAB, LPZ, and in the CR resulting from an SGTR are well within the applicable limits. It is confirmed that the simplified calculations with a 30-minute break flow and release duration used in the licensing basis analysis produce significantly higher dose results compared to doses calculated with transient data (from Section 3.3) obtained modeling conservative operator response times and break flow continuing beyond 30 minutes.

Table 9
Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture

RCS Iodine Activity	
1. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gm DE I-131. These are 60 times the values given in Table 10 which are based on 1.0 μCi/gm DE I-131.
2. Accident-Initiated Spike	Initial primary coolant iodine activities based on 1.0 μCi/gm DE I-131. The iodine appearance rates for the accident-initiated spike are 335 times the equilibrium appearance rates which are given in Table 10. The spike continues until 8 hours from the start of the event.
Noble Gas Activity	Primary coolant noble gas activities based on operation with defects in fuel producing 1% of core power. Values are given in Table 10.
RCS Bromine and Alkali Metal Activity	Primary coolant Bromine and alkali metal activities based on operation with defects in fuel producing 1% of core power. Values are given in Table 10.
Secondary Coolant System Activity	Initial secondary coolant iodine activity based on 0.1 μCi/gm DE I-131. SG halogen and alkali metal activity concentrations are assumed to be 1/10th of the RCS activity concentrations, reflecting the ratio of the Technical Specifications limits on DE I-131. No noble gases are contained in the secondary coolant. Values are given in Table 10.
Iodine Chemical Fractions	97% elemental, 3% organic, no particulates (no impact on analysis since filter efficiencies credited are the same for all forms of iodine)
RCS Mass	Hand calc: Constant at 3.4E5 lbm Transient: Constant at 3.4E5 lbm (Table 8)
Intact SGs Mass	Hand calc: Constant at 2.12E5 lbm Transient: Constant at 1.9E5 lbm (Table 8)
Ruptured SG Mass	Hand calc: Constant at 1.06E5 lbm Transient: Constant at 9.6E4 lbm (Table 8)

Table 9 (continued) Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture	
Transient Timing	
Reactor Trip, SI , LOOP	Hand calc: 385 sec Transient: 76 sec (Table 5)
Break Flow Flashing Stops	Hand calc: 1800 sec Transient: 2067 sec (Table 5)
Break Flow Terminates	Hand calc: 1800 sec Transient: 3411 sec (Table 5)
Intact SG Releases Terminated	24 hours
Ruptured SG Transient Release Data	
Rupture Flow** Pre-Trip Post-Trip Until Flashing Stops Post-Trip After Flashing Stops	 Hand calc: 19,400 lbm Transient: 3600 lbm (Table 6 and Figure 6*) Hand calc: 73,500 lbm Transient: 94,240 lbm (Table 6 and Figure 6*) Hand calc: 0 lbm Transient: 37,880 lbm (Table 6 and Figure 6*)
Flashed Rupture Flow Pre-Trip Post-Trip	 Hand calc: 3,880 lbm Transient: 610 lbm (Table 6 and Figure 9*) Hand calc: 9,188 lbm Transient: 5,120 lbm (Table 6 and Figure 9*)
Steam Releases Pre-Trip Post-Trip Until Break Flow Stops	 1,310 lbm/sec Hand calc: 56,800 lbm post trip Transient: 73,740 lbm (Table 7 and Figure 7*)

Table 9 (continued) **Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture**	
Intact SGs Transient Release Data***	
Primary to Secondary Leakage	1 gpm (8.34 lbm/min)
Steam Releases Pre-Trip Post-Trip Until 2 hours 2 to 8 hours 8 to 24 hours	 2,620 lbm/sec Hand calc: 381,400 lbm Transient: 348,300 lbm until break flow termination (Table 7 and Figure 7*), 256,400 from break flow termination until 2 hours 924,900 lbm 1,200,000 lbm
Iodine Partition Coefficients	
Condenser	Not modeled
Steam Release from SGs	100
Flashed Break Flow Release from Ruptured SG	1.0
Atmospheric Dispersion Factors	Values are given in Table 11
Breathing Rate EAB LPZ	 3.5E-4 m^3/sec for all time intervals 3.5E-4 m^3/sec until 8 hours 1.8E-4 m^3/sec from 8 to 24 hours
Control Room Modeling	See Table 13

* The analysis conservatively added 10% to the mass transfer data presented in this figure.
** This is the total flow through the break and includes the flashed flow listed separately.
*** Data listed in total for both intact SGs.

Table 10
Specific Activities in the Primary Coolant and Associated Iodine Appearance Rates and Specific Activities in the Secondary Coolant

Nuclide	RCS Concentration Based on 1.0 μCi/gm DE I-131 (μCi/gm)	Equilibrium Appearance Rate (Ci/sec)	RCS Concentration Based on Operation with Defects in Fuel Producing 1% of Core Power (μCi/gm)	SG Concentration (μCi/gm)
I-131	0.782	7.21E-3	-	0.0782
I-132	1.268	3.33E-2	-	0.1268
I-133	1.268	1.39E-2	-	0.1268
I-134	0.279	1.53E-2	-	0.0279
I-135	0.862	1.29E-2	-	0.0862
Br-83	-	-	0.089	0.0089
Br-84	-	-	0.042	0.0042
Cs-134	-	-	4.4	0.44
Cs-136	-	-	4.5	0.45
Cs-137	-	-	2.1	0.21
Cs-138	-	-	0.97	0.097
Rb-88	-	-	6.5	0.65
Kr-85m	-	-	1.8	-
Kr-85	-	-	7.9	-
Kr-87	-	-	1.1	-
Kr-88	-	-	3.2	-
Xe-131m	-	-	3.4	-
Xe-133m	-	-	19.0	-
Xe-133	-	-	290	-
Xe-135m	-	-	0.52	-

Table 10 (continued)
Specific Activities in the Primary Coolant and Associated Iodine Appearance Rates and Specific Activities in the Secondary Coolant

Nuclide	RCS Concentration Based on 1.0 µCi/gm DE I-131 (µCi/gm)	Equilibrium Appearance Rate (Ci/sec)	RCS Concentration Based on Operation with Defects in Fuel Producing 1% of Core Power (µCi/gm)	SG Concentration (µCi/gm)
Xe-135	-	-	8.6	-
Xe-138	-	-	0.64	-

Table 11 **Atmospheric Dispersion Factors**			
Time Period	Exclusion Area Boundary (sec/m^3)	Low Population Zone (sec/m^3)	Control Room (sec/m^3)
0 to 2 hours	1.24E-4	5.06E-5	1.51E-3
2 to 8 hours	1.24E-4	2.42E-5	1.17E-3
8 to 24 hours	1.24E-4	1.68E-5	5.75E-4

Table 12 Dose Conversion Factors		
Nuclide	Reference 11 EDE Dose Conversion Factor - Cloudshine ($Sv\text{-}m^3/Bq\text{-}sec$)	Reference 10 CEDE Dose Conversion Factor - Inhaled (Sv/Bq)
I-131	1.82E-14	8.89E-09
I-132	1.12E-13	1.03E-10
I-133	2.94E-14	1.58E-09
I-134	1.30E-13	3.55E-11
I-135	8.294E-14[1]	3.32E-10
Br-83	3.82E-16	2.41E-11
Br-84	9.41E-14	2.61E-11
Cs-134	7.57E-14	1.25E-08
Cs-136	1.06E-13	1.98E-09
Cs-137	2.88E-14[2]	8.63E-09
Cs-138	1.21E-13	2.74E-11
Rb-88	3.36E-14	2.26E-11
Kr-85m	7.48E-15	-
Kr-85	1.19E-16	-
Kr-87	4.12E-14	-
Kr-88	1.02E-13	-

[1] The value listed for I-135 is the DCF for I-135 plus the DCF for its daughter product Xe-135m adjusted by the branching fraction of 0.154.

[2] The value listed for Cs-137 is the DCF for its daughter product Ba-137m.

Table 12 (continued) **Dose Conversion Factors**		
Nuclide	Reference 11 EDE Dose Conversion Factor - Cloudshine (Sv-m^3/Bq-sec)	Reference 10 CEDE Dose Conversion Factor - Inhaled (Sv/Bq)
Xe-131m	3.89E-16	-
Xe-133m	1.37E-15	-
Xe-133	1.56E-15	-
Xe-135m	2.04E-14	-
Xe-135	1.19E-14	-
Xe-138	5.77E-14	-

Table 13 Control Room Modeling	
Transition from Normal Mode Ventilation, to Emergency Mode	The CR ventilation emergency mode is initiated at 0.5 hours after the start of the accident.*
CR Volume	226,040 ft^3
CR Unfiltered In-Leakage Normal Mode Emergency Mode	 217 cfm 243 cfm
CR Unfiltered Makeup Flow Normal Mode Emergency Mode	 1,291 cfm 0 cfm
CR Filtered Makeup Flow Normal Mode Emergency Mode	 0 cfm 1,265 cfm
CR Filtered Recirculation Flow Normal Mode Emergency Mode	 0 cfm 19,125 cfm
CR Filter Efficiency	95% for elemental and organic iodines and particulates
CR χ/Q	Values are given in Table 11
CR Breathing Rate	3.5E-4 m^3/sec
CR Occupancy Factors	1.0 for the first day 0.6 from 1 to 4 days 0.4 after 4 days

* The conservative time has been retained to be consistent with the approved analysis (Reference 1). In addition, sensitivity calculations were performed crediting the emergency mode of operation 60 seconds after the SI signal.

Table 14
SGTR Radiological Consequences Analysis Results

Scenario	Location	Hand Calc TEDE Dose (rem)*	Transient TEDE Dose (rem)*	TEDE Dose Limit (rem)
Pre-Accident Iodine Spike	Exclusion Area Boundary (0 to 2 hours)	0.68	0.34	25
	Low Population Zone (0 to 24 hours)	0.28	0.14	25
	Control Room (0 to 30 days) CR ventilation emergency mode is initiated at:			5
	30 minutes	1.30	0.53	
	SI Actuation + 60 seconds	0.37	0.11	
Accident-Initiated Iodine Spike	Exclusion Area Boundary (0 to 2 hours)	0.31	0.18	2.5
	Low Population Zone (0 to 24 hours)	0.14	0.088	2.5
	Control Room (0 to 30 days) CR ventilation emergency mode is initiated at:			5
	30 minutes	0.50	0.22	
	SI Actuation + 60 seconds	0.13	0.06	

* Approximately 5% margin was added to the total calculated dose to arrive at the dose reported. This margin is included to allow flexibility in addressing minor impacts on the dose analysis without requiring a reanalysis or changes in the reported dose.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The Margin to Steam Generator Overfill Analysis was performed to demonstrate that the secondary side of the ruptured SG did not completely fill with water. The available secondary side volume of a single SG is 5543 cubic feet. Margin-to-overfill is demonstrated provided the transient calculated SG secondary side water volume is less than this value. This volume does not take any credit for the volume of the nozzle or any of greater than 450 cubic feet of steam piping which would have to be filled with water prior to water being released to the environment from the ruptured SG.

For the Radiological Consequences Analysis, the offsite TEDE dose limits for the two iodine spike scenarios are specified in RG 1.183 (Reference 8) and Standard Review Plan (SRP) 15.0.1 (Reference 12). The doses at the EAB and the LPZ for an SGTR with an assumed pre-accident iodine spike must meet the 10 CFR 50.67 limit of 25 rem TEDE, while the EAB and LPZ doses for an SGTR with an assumed accident-initiated iodine spike must meet the limit of 2.5 rem TEDE specified in RG 1.183. The EAB doses are calculated for the limiting two hours. The LPZ doses are calculated up to the time releases are terminated, which is 24 hours.

The CR dose limit of 5 rem TEDE is specified in SRP 6.4 (Reference 13) based on 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. The CR doses are calculated for 30 days.

4.2 PRECEDENCE

VCSNS's proposed changes use the same general approach with similar analyses to the approved license amendments for Point Beach Nuclear Plant Units 1 and 2, Turkey Point Units 3 and 4, and Prairie Island Nuclear Generating Plant Units 1 and 2. The applicable references for the license amendments are provided below:

NRC Letter, Point Beach Nuclear Plant (PBNP), Units 1 and 2 – Issuance of License Amendments Regarding Extended Power Uprate (TAC NOS. ME1044 and ME1045), dated May 3, 2011 [ML110450159 and ML111170513]

NRC Letter, Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Extended Power Uprate (TAC NOS. ME4907 and ME4908), dated June 15, 2012 [ML11293A359 and ML11293A365]

NRC Letter, Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: Adoption of Alternative Source Term Methodology (TAC NOS. ME2609 and ME2610), dated January 22, 2013 [ML112521289]

In the license amendment requests that were approved, the SGTR analysis for input to dose maintained the old licensing basis 30-minute hand calculation and was supplemented by (1) a calculation to show that with respect to dose inputs the 30-minute hand calculation is

conservative compared to a calculation modeling actual operator actions and leading to greater than 30 minutes of break flow and (2) a margin-to-overfill analysis that followed WCAP-10698-P-A but with no single failure (consistent with the plants' original licensing bases) and with changes to address NSAL-07-11. Although the inputs and assumptions used for the approved amendments are different, the same general approach with similar analyses was used.

4.3 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Pursuant to 10 CFR 50.92, VCSNS has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration, since the proposed change satisfies the criteria in I0 CFR 50.92(c). These criteria require that the operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (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. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

1.0 Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

Response: No.

The proposed change, to adopt a new analytical method to evaluate the effects of a SGTR, does not affect any accident initiators or precursors since there is no physical change to plant systems, structures and components or manner in which they are operated during normal operation. As such, the proposed change does not increase the probability of an accident.

The ability of operators to mitigate the consequences of an accident is also not diminished as there is no impact on the design of mitigating plant systems that would reduce their design capability or increase their failure probability during normal operation or accident conditions.

The present methodology for calculating mass transfer (i.e., from the RCS to the secondary side via the failed SG tube) for input to the radiological consequences of a postulated SGTR is conservative when compared with results from the new methodology. As such, the existing licensing basis methodology for calculating mass transfer will be retained. The calculated doses for the SGTR event for use in the FSAR will be updated to reflect the results of the updated calculations with the reported doses to include 5 percent margin. Although slightly higher than the current analyses of record, the updated doses are well within regulatory limits and the increases are not more than minimal. Consistent with VCSNS's current licensing basis, the dose calculations conform to the guidance presented in 10 CFR 50.67, RG 1.183, and Standard Review Plan Section 15.0.1.

The use of this previously approved methodology (WCAP-10698-P-A) more accurately calculates the plant response to an SGTR event. The improved accuracy of the new methodology provides valuable information related to operator actions and associated timing. Such accurate transient response information enables enhancements to be made to the emergency operating procedures (EOPs) and allows future changes to be more effectively assessed for impact.

Therefore, the proposed change does not involve a significant increase in the consequences or probability of occurrence of an accident previously evaluated.

2.0 Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not impact the design of affected plant systems, involve a physical alteration to the systems, or change to the way in which systems are currently operated, such that previously unanalyzed SGTRs would now occur. Since the design function and mode of operation of SSCs in the facility prior to a postulated accident are unchanged, the change to adopt a new analytical method to evaluate the effects of a tube rupture does not introduce any new malfunctions. Its use is beneficial in that it allows for a more accurate prediction of the plant response following a postulated SGTR to determine the time available for operator actions to prevent overfilling the affected SG. Thus, the proposed change does not affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

3.0 Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The approval of the proposed change will not result in any modifications to affected plant systems that would reduce their design capabilities during normal operating and accident conditions. By using the WCAP-10698-P-A methodology, a more accurate SGTR response is calculated. The improved understanding of the transient response enables enhancements to the EOPs, which provide further assurance the SSCs required for accident mitigation are available and that required actions can be accomplished in a time frame to prevent overfill of a ruptured SG.

The SGTR dose consequences to be reported in the FSAR are well within the acceptance criteria presented in 10 CFR 50.67, RG 1.183, and SRP 15.0.1. Given this, there is no significant reduction in a margin of safety.

Conclusion

On the basis of the above, VCSNS has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it: (1) does not involve a significant increase in the probability or

consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

4.4 CONCLUSIONS

- SG overfill will not occur for a design basis SGTR.
- The doses at the EAB, LPZ, and in the CR resulting from a design basis SGTR are well within the applicable limits.
- The simplified calculations with a 30-minute break flow and release duration used in the licensing basis analysis produce significantly higher dose results compared to doses calculated with transient data modeling validated operator response times with break flow continuing beyond 30 minutes.

5.0 ENVIRONMENTAL CONSIDERATION

VCSNS has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

VCSNS has determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). 10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license or change in licensing basis for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed change would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. VCSNS has evaluated the proposed change and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

<u>Basis</u>

As demonstrated in the *No Significant Hazards Consideration Evaluation*, the proposed change to the VCSNS licensing basis does not involve a significant hazards consideration.

The proposed change does not involve a physical alteration of the plant or change in methods governing normal plant operation. Instead, the change provides additional analyses of a SGTR accident which explicitly models operator responses and quantifies their impact on the potential

for steam generator overfill and offsite and control room doses. The new transient calculations supplement the Virgil C. Summer licensing basis analysis by demonstrating margin to steam generator overfill and providing input to dose analyses that confirm that the licensing basis mass transfer input to the SGTR dose analysis is conservative. Given that the doses are well within regulatory limits, the proposed change to the VCSNS licensing basis does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

There is no significant increase in individual or cumulative occupational radiation exposure. The proposed change does not involve any physical alteration of the plant or change in methods governing normal plant operation.

Conclusion

The proposed change does not affect the design or normal operation of the facility; rather, given a postulated SGTR, the proposed change will supplement the VCSNS licensing basis by demonstrating margin to steam generator overfill and providing input to dose analyses that confirm that the licensing basis mass transfer input to the SGTR dose analysis is conservative.

On the basis of the above, VCSNS has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 51.22(c)(9), in that it does not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

6.0 REFERENCES

1. "Virgil C. Summer Nuclear Station, Unit No. 1, Issuance of Amendment Regarding Alternative Source Term Implementation (TAC No. ME0663)," October 2010. (ADAMS Accession Number ML102160020).

2. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987.

3. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," March 1986.

4. WCAP-16948-P, "Clarifications for the Westinghouse Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology," December 2008.

5. NSAL-07-11, "Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology," November 2007.

6. "Point Beach Nuclear Plant (PBNP), Units 1 and 2 - Issuance of License Amendments Regarding Use of Alternate Source Term (TAC Nos. ME0219 and ME0220)," April 14, 2011. (ADAMS Accession Number ML110240054).

7. TB-07-6, "Credited Relief Capacity of Atmospheric Steam Relief System," May 2007.

8. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

9. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," December 1997, and its Supplements 1 and 2 dated June 1999 and October 2002, respectively.

10. K. F. Eckerman et al., "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, Environmental Protection Agency, September 1988.

11. K. F. Eckerman and J. C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, EPA-402-R-93-081, Environmental Protection Agency, September 1993.

12. NUREG-0800, Standard Review Plan, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000.

13. NUREG-0800, Standard Review Plan, Section 6.4, "Control Room Habitability System," Revision 3, March 2007.

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
DOCKET NO. 50-395
OPERATING LICENSE NO. NPF-12

ATTACHMENT 1

PLANT SPECIFIC INPUT TO SUPPORT USE OF WCAP-10698-P-A

Nomenclature

CFR	Code of Federal Regulations
EF	Emergency Feedwater
EOP	Emergency Operating Procedure
ESFAS	Emergency Safety Features Actuation System
FW	Feedwater Isolation Valve
IA	Instrument Air
MCB	Main Control Board
MDEFP	Motor Driven Emergency Feedwater Pump
MOV	Motor Operated Valve
MS	Main Steam
PZR	Pressurizer
PORV	Power Operated Relief Valve
RB	Reactor Building
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RMA	Radiation Monitor - Atmospheric
RMG	Radiation Monitor - Gaseous
RML	Radiation Monitor - Liquid
RPS	Reactor Protection System
RSG	Ruptured Steam Generator
SER	Safety Evaluation Report
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SRP	Standard Review Plan
SS	Secondary Side
T_{AVG}	Average RCS Coolant Temperature
T_{COLD}	RCS Loop Cold Leg Temperature
T_{HOT}	RCS Loop Hot Leg Temperature
TDEFP	Turbine Driven Emergency Feedwater Pump
VCSNS	Virgil C. Summer Nuclear Station
1E	Class 1E per
1σ	One Standard Deviation

Section D, "Plant Specific Submittal Requirement," of Enclosure 1 to the safety evaluation for WCAP-10698-P-A (Enclosure of LAR, Reference 2), states that certain plant specific inputs shall be provided when referencing the WCAP for licensing actions. The five items described in Section D of the SER are provided below with responses following each item.

1. **Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design-basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention, using design-basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.**

The required operator action times used in the VCSNS specific SGTR margin-to-overfill are summarized in Section 3.2.2.2 and Table 3 of the enclosure to this license amendment request (LAR). These have been shown to be realistic via simulator exercises to assure the operators could mitigate the accident within the period of time compatible with overfill prevention. Sample results are shown in Table 15; as indicated, the required times were met by all operating crews and the mean times plus one standard deviation are also adequate to preclude overfill.

The operator action times used in the VCSNS analysis will be added to the plant's operator requalification program which currently requires both classroom and simulator SGTR training for all licensed operators.

All plant modifications and/or major changes to the EOPs are assessed for impact to assure the EOPs remain compatible with the plant design and licensing basis, including SGTR scenarios. Major changes to EOPs are also assessed for impact on time critical actions and, when adverse impacts are suspected, the ability of the operator to perform within acceptable time limits is revalidated.

2. **A site-specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the Guidance in Reference (1).**

As discussed in the enclosure of this LAR (Section 3.4), a site-specific SGTR radiological consequence analysis was performed following the guidance provided in Regulatory Guide (RG) 1.183 (Enclosure of LAR, Reference 8) and using the RADTRAD code, Version 3.03 (Enclosure of LAR, Reference 9). The calculations determined the doses based on a pre-accident iodine spike and on an accident-initiated iodine spike. Consistent with VCSNS's original and current licensing basis, an additional failure within the mitigating equipment set was not assumed.

Two sets of SGTR mass release data were considered. One set of mass transfer data is consistent with that used in the licensing basis SGTR dose analysis, based on the simplified

30-minute break flow and release duration. The second set of mass transfer data is that presented in Section 3.3 of the enclosure of this submittal, which explicitly accounts for operator action in accordance with the VCSNS EOPs with times in excess of 30 minutes.

As shown in Table 14 in the enclosure of this LAR, large margins to offsite and control room radiological limits are maintained. A comparison of the calculated doses also demonstrate that the mass release data used in the licensing basis SGTR dose analysis is conservative even though it does not consider break flow continuing beyond 30 minutes.

3. An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.

The margin to SG overfill analysis following a SGTR determined that an overfilled condition would not occur. This conclusion is supported by operator action timing and plant simulator runs. However, from a defense in depth perspective, an assessment was performed for the main stream lines up to the main stream isolation valves for water filled conditions.

Should overfill occur, static loads would evolve due to the weight of the water within the pipes, temperature differences between the piping and contained fluid and internal pressure. In addition, at the time overfill could occur during a SGTR, sufficient time will have elapsed for the operators to terminate EF flow to the ruptured SG and the addition of water to the SG will be solely from the break flow. Break flows would also decrease as action are taken to cooldown and equalize primary and secondary pressures. Therefore, the long term fill process is expected to be slow, and conditions conducive to water hammer (i.e., rapid condensation between water and steam, sudden interruption of high velocity liquid flow or entrainment of water in a steam filled line) would not exist. Consequently, only deadweight, thermal and pressure stresses/loads for water filled conditions were considered. The assessment indicates that the piping and supports can accommodate these loads and remain within code allowable, thus ensuring that steam line structural integrity will be maintained.

4. A list of systems, components and instruments which are credited for accident mitigation in the plant specific SGTR EOP(s). Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade. For non-safety grade systems and components state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured SG identification, provide the effect on the duration of the accident.

As indicated in the enclosure of this LAR, a design basis SGTR leads to a reactor trip and safety injection (SI) actuation and would thus require entry into EOP-1.0, REACTOR TRIP/SAFETY

INJECTION ACTUATION. Following completion of immediate actions, additional actions are taken for SGTR accident mitigation. These actions focus on five key aspects:

- Identifying the ruptured SG,
- Isolating the ruptured SG,
- Cooling down the reactor coolant system (RCS),
- Depressurizing the RCS, and
- Terminating SI flow to the RCS.

A list of the systems, components and instruments utilized by the operators for SGTR mitigation under design basis conditions (i.e., SGTR plus a loss of offsite power), is provided in Table 16. Emphasis is placed on identifying the indications and equipment that the operator will utilize and/or manipulate following completion of EOP-1.0 immediate actions. Mitigation is assumed to be achieved when the pressure in the RCS and the ruptured SG has been equalized and SI terminated. Actions and equipment required for long term recovery are not addressed in Table 16 since the potential for SG overfill and/or significant releases no longer exist.

A number of the items listed in Table 16 must be utilized because a concurrent loss of offsite power is assumed. This includes use of the main steam PORVs on the intact SGs (i.e., instead of condenser dumps) to cool down the RCS, use of pressurizer PORVs (i.e., instead of normal pressurizer spray) for depressurizing the RCS and reliance on backup air supplies.

The station's compressed air systems are discussed in Section 9.3.1 of the FSAR and include the Station Service Air, Instrument Air, and Reactor Building Instrument Air Systems. The major components are shown in FSAR Figures 9.3-1 through 9.3-3a. The air systems are non-safety except for the associated Reactor Building penetrations and process piping between and including containment isolation valves. As discussed therein:

> The Station Service and Instrument Air System are supplied by two (2) electric motor driven, full capacity compressors (XAC-3A/B) located in the turbine room. Each air compressor motor receives electric power from a different non-safety related 480 volt bus and is controlled locally or from the control room. Typically, one air compressor is in operation, and the other is in standby with auto-start on low air pressure.
>
> A normally open valve interconnects the Station Service and Instrument Air system with a backup compressor (XAC-12), which acts as a third standby compressor. XAC-12 receives electric power from a 480 volt diesel backed bus (XSW1DB1) which is isolated in the event of loss of offsite power or safety injection. In the event of a loss of offsite power, this compressor can be manually loaded onto the diesel bus and restarted following a local reset.
>
> A Diesel Driven Air Compressor (XAC-14) provides an emergency backup air source when both Instrument Air Compressors and their backup are not available. XAC-14 is located in the yard within the protected area. It is a rotary screw air compressor which runs in either a "loaded" condition or an "unloaded" condition. XAC-14 is automatically

started in the event of a loss of power to the in-plant compressors or loss of instrument air header pressure.

Lastly, during normal operation, the RB is served by two full capacity air compressors (XAC-4A/B) located in the penetration access area. The air compressors take suction from the Reactor Building and discharge to air receivers. Typically, one of the compressors is in operation and the other in standby with auto-start on low air pressure. Both are driven by electric motors that receive power from a different 480 volt, nonessential bus and can be controlled locally or from the control room.

For SGTR mitigation, the key air-operated valves of interest for mitigation are the pressurizer PORVs (i.e., for RCS depressurization), SG PORVs (i.e., for RCS cooldown), EF flow control valves (i.e., for isolation of the ruptured SG and level control in the intact SGs). In the longer term, the auxiliary pressurizer spray valve (i.e., for backup RCS depressurization), and the charging/letdown valves for RCS inventory control are also utilized for event recovery. A brief overview of each follows:

<u>Pressurizer PORVs (PCV-444B, 445A-RC)</u> – these normally closed, air operated valves are located inside containment and fail closed on loss of air or power. However, the valves are safety related / seismic category 1 / redundant, have backup seismic category 1 air supply accumulators, and can be operated manually from the MCB using a 1E circuit. The SGTR analysis assumes at least one pressurizer PORV can be operated from the Main Control Board (MCB).

<u>SG PORVs (IPV-2000, 2010, 2020)</u> – these normally closed, air operated valves are located outside containment. They are safety related / seismic category 1 and are provided with safety related / seismic category 1 remote operators. Remote control circuits are, however, control grade. The valves fail closed on loss of air or power. Backup air accumulators are not provided. The SGTR analysis assumes the PORVs on the intact SGs can be operated from the MCB to facilitate RCS cooldown. The valves are equipped with hand-wheels to permit local operation, if needed.

<u>Auxiliary Spray Valve (XVT-8145)</u> – this normally closed, air operated valve is located inside containment. It is safety related / seismic category 1 and is provided with a safety related / seismic category 1 remote operator. Remote control circuits are, however, control grade. The valve fails closed on loss of air or power. A backup air accumulator is not provided. This valve is not explicitly modeled in the SGTR analysis. Its use is covered in the EOPs as an alternative action to depressurize the RCS in the long term (i.e., when normal pressurizer spray is not available). Its use requires normal letdown and charging to be in-service.

<u>Emergency Feedwater Flow Control Valves (IFV-3531, 3541, 3551, 3536, 3546, 3556-EF)</u> – these normally open, air operated valves are located outside containment and fail open on loss of air. They are safety related / seismic category 1 and are provided with safety related / seismic category 1 remote operators and 1E control circuits. They

also have backup safety related / seismic category 1 air accumulators. Within the SGTR analysis (per EOP-4.0), it is assumed that the operator can (1) isolate Emergency Feedwater to the ruptured SG by closing the associated Emergency Feedwater flow control valves and (2) regulate Emergency Feedwater addition to the intact SG to maintain SG level. The valves are equipped with hand wheels to permit local operation, if needed.

Letdown Valves – a number of normally open, air operated valves are in the letdown flow path. Key valves include LCV-459/460, XVT-8149A/B/C, and containment isolation valve XVT-8152. These valves are safety related / seismic category 1 and fail closed on loss of instrument air. They are provided with safety related / seismic category 1 remote operators; however, except for XVT-8152, the remote control circuits are non-1E. Backup accumulators are not provided. Letdown is not explicitly modeled in the SGTR analysis. The thermal hydraulic analysis is terminated following equalization of pressure between the RCS and the ruptured SG; thereafter, it is assumed that RCS inventory would be controlled. The EOPs call for restoration of normal charging and letdown (if possible) in preparation for further RCS cooldown and depressurization to go on the RHR system.

Charging Valves (FCV-122 & XVT-8146): these normally open, air operated valves fail open on loss of air. FCV-122 is located outside containment whereas XVT-8146 is located inside containment. They are safety related / seismic category 1 and are provided with safety related / seismic category 1 remote operators. Remote control circuits are, however, non-1E; and, backup accumulators are not provided. Charging is not explicitly modeled in the SGTR analysis. The thermal hydraulic analysis is terminated following equalization of pressure between the RCS and the ruptured SG; thereafter, it is assumed that RCS inventory would be controlled. The EOPs call for restoration of normal charging and letdown (if possible) in preparation for further RCS cooldown and depressurization to go on RHR.

As discussed above, redundant, non-safety compressors are available for mitigation of and recovery from a SGTR event with a concurrent loss of offsite power: the Station Service and Instrument Air system backup compressor (XAC-12) and the Diesel-Driven Air Compressor (XAC-14). The most immediate indication of a problem with IA would be a main control board annunciator alarm (audible/visual) on low header air pressure. Actions for restoration of IA are specified in the alarm response procedure, including verification that the Diesel Driven Air Compressor (XAC-14) automatically starts. To minimize the potential for any delays due to a loss of air, XAC-14 is manually started following entry into the SGTR EOP.

The following radiation monitors are available for identification of a SGTR event:

- Steam Line Radiation Monitors (RMG-19A/B/C)
- Condenser Exhaust - Gas Atmospheric (RMA-9)
- Steam Generator Blowdown - Liquid (RML-3)
- SG Blowdown Discharge Monitor - Liquid (RML-10)

All of the monitors listed above are non-safety grade and are specifically listed in SGTR procedures for identification of a SGTR event. These instruments are described in Section 11.4.2 of the FSAR and are maintained and tested in accordance with VCSNS Technical Specifications or Offsite Dose Calculation Manual requirements. To ensure reliable indication/detection in the event of a SGTR, the monitors are also scoped into the VCSNS Maintenance Rule program in accordance with 10CFR50.65, and monthly source checks are performed on the atmospheric and liquid monitors with results trended to detect detector degradation or high background.

The SGTR EOP allows identification of the ruptured SG by any of the following:

- High steam line radiation as detected by RMG-19A/B/C
- An unexpected rise in SG narrow range level
- Local hand held radiation monitor readings of the individual SG blow down lines
- Hand held radiation monitor reading of individual SG liquid samples

The most limiting accident with respect to SG overfill is the complete severance of one SG tube. The high primary to secondary flow that results from this design basis accident allows the ruptured SG to be promptly identified by the operators due to high steam line radiation levels or the unexpected rise in SG narrow range level. The other methods provide a longer-term confirmatory indication of the ruptured SG.

5. **A survey of plant primary and "balance-of-plant" systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the difference on the margin of overfill should be provided.**

This item is not applicable as the bounding plant analysis as described in WCAP-10698-P-A (Enclosure of LAR, Reference 2) is not used. Rather, this submittal makes use of the WCAP-10698-P-A methodology (and associated computer code) to supplement the VCSNS SGTR analyses with a VCSNS-specific margin to SG overfill analysis. The analysis uses plant specific input values and, consistent with the plant's licensing basis, no single failure is assumed. Since the supplemental margin to SG overfill analysis is plant specific, the WCAP-10698-P-A bounding plant analysis was not used, or needed.

Table 15 Simulator Time Line Results

Action	Time (Minutes)									
	Analysis Value per Table 3	'A' Shift	'B' Shift	'C' Shift	'D' Shift	'E' Shift	Mean	1σ	Mean + 1σ	Margin
Time to isolate EFW to ruptured SG after reactor trip, (min)	**6**	5.6	3.37	4.93	2.27	5.42	4.318	1.462	5.780	0.220
Time to initiate cooldown (intact SG PORVs full open) from time of trip, (min)	**15**	12.39	13.43	12.38	12.4	10.12	12.144	1.596	13.740	1.260
Time to initiate RCS depressurization (Pzr PORV full open) from time of cooldown termination (SG PORV begins to close), (min)	**4**	2.67	2.98	1.7	2.27	2.4	2.404	0.780	3.184	0.816
Time to terminate SI flow (8801A/B full closed) from time of RCS depressurization termination (Pzr PORV begins to close), (min)	**4**	1.83	1.58	1.97	1.95	1.55	1.776	0.925	2.701	1.299

Table 16 Equipment Credited for SGTR Mitigation in the VCSNS Emergency Operating Procedures [EOP(s)]

Item	Safety Grade (Y or N)	Comment	SGTR Function	Functional Remark
Pressurizer Level LI-459 LI-460 LI-461	Y	Loops provide input to the RPS. MCB indicators are RG 1.97, Category 1. (FSAR Figure 5.1-1, Sht. 2)	SGTR Identification RCS Depressurization SI Termination Criteria	1, 2 3 4
Pressurizer Pressure PI-455 PI-456 PI-457	Y	Loops provide input to the RPS and ESFAS. MCB PI-457 is RG 1.97, Category 1. (FSAR Figure 5.1-1, Sht. 2)	SGTR Identification SI Actuation	1, 2 5
Wide Range RCS Pressure PI-402 PI-403	Y	Loops provide input to the Core Subcooling Monitor. MCB indicators are RG 1.97, Category1. (FSAR Figure 5.1-1, Sht.1)	RCS Depressurization SI Termination Criteria Pressure	6 4
Wide Range RCS Tcold TI-410 TI-420	Y	Loops provide input to the Core Subcooling Monitor. MCB indicators are RG 1.97, Category 1. (FSAR Figure 5.1-1, Sht.1)	General	
Wide Range RCS Thot TI-413 TI-423	Y	Loops provide input to the Core Subcooling Monitor. MCB indicators are RG 1.97, Category 1. (FSAR Figure 5.1-1, Sht.1)	General	

Table 16 Continued - Equipment Credited for SGTR Mitigation in the VCSNS Emergency Operating Procedures [EOP(s)]

Item	Safety Grade (Y or N)	Comment	SGTR Function	Functional Remark
Feedwater Flow FI-476 FI-486 FI-496	N	Loops provide input to RPS. MCB indicators are RG 1.97, Category 3. (FSAR Figure 10.4-12)	RSG Identification	1, 7
SG Narrow Range Level LI-474/484/494 LI-475/485/495 LI-476/486/496	Y	Loops provide input to the RPS. MCB indicators are RG 1.97, Category 1. Trend recorder for one level channel per SG is also available (LI-476/486/496) (FSAR Figure 10.3-1)	RSG Identification SG Level Control	1, 7, 8 9
Steam Line Pressure PI-474/475/476 PI-484/485/486 PI-494/495/496	Y	Loops provide input to the RPS. MCB indicators are RG 1.97, Category 1. (FSAR Figure 10.3-1)	RCS Cooldown	10
EF Flow FI-3561 FI-3571 FI-3581	Y	MCB indicators provide EF flow to each SG and are RG-1.97, Category 1. (FSAR Figure 10.4-6)	RCS Heat Removal SI Termination Criteria	11 4

Table 16 Continued - Equipment Credited for SGTR Mitigation in the VCSNS Emergency Operating Procedures [EOP(s)]

Item	Safety Grade (Y or N)	Comment	SGTR Function	Functional Remark
Core Exit Thermo-couples TC-01 through 51	Y[12]	Output is displayed on plant computer. A backup display for 16 (4 per quadrant) is available on the Core Subcooling Monitor. (FSAR Section 7.5.5)	Monitor RCS Cooldown	12
RCS Subcooling Monitor TM-499A(B)	N[13]	System displays saturation margin and core exit temperature. Subcooling, based on core exit thermocouples, is a RG-1.97 category 2 indication. (FSAR Section 7.5.5)	SI Termination Criteria	4, 13
SG PORVs (air operated) PV-2000 PV-2010 PV-2020	N[16]	See Attachment 1, Item 4 for SG PORV attributes. ESF Monitor Light on MCB provides RG-1.97 valve position indication. (FSAR Figure 5.1-1, Sht.2)	RSG Isolation RCS Cooldown	14 15, 16
EF Flow Control Valve (air operated) IFV-3531 IFV-3541 IFV-3551 IFV-3536 IFV-3546 IFV-3556	Y	See Attachment 1, Item 4 for EF flow control valve attributes. (FSAR Figure 10.4-6)	EF Control RSG Isolation	17 18

Table 16 Continued - Equipment Credited for SGTR Mitigation in the VCSNS Emergency Operating Procedures [EOP(s)]

Item	Safety Grade (Y or N)	Comment	SGTR Function	Functional Remark
FW Isolation Valve XVG-1611A(B)(C)	Y	Valve status (open/closed) verified using the MCB control switch light, a RG-1.97 Category 2 indication. (FSAR Figure 10.4-12)	FW Isolation	19, 20
TDEFP Steam Supply Flow Control Valve (air operated) IFV-2030	Y	Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication. (FSAR Figure 10.3-1)	RSG Isolation	21
SG Blowdown Isolation Valve (air operated) XVG-503A(B)(C)	Y	Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication. (FSAR Figure 10.4-13)	RSG Isolation	23
MS Drain Isolation Valve (air operated) XVT-2843A(B)(C) XVT-2877A(B)	Y	Valve status (open/closed) is verified using the MCB control switch light. (FSAR Figure 10.3-1)	RSG Isolation	24
MS Isolation Valves XVM-2801A(B)(C)	Y	Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication. (FSAR Figure 10.3-1)	RSG Isolation	25

Table 16 Continued - Equipment Credited for SGTR Mitigation in the VCSNS Emergency Operating Procedures [EOP(s)]

Item	Safety Grade (Y or N)	Comment	SGTR Function	Functional Remark
MS Isolation Bypass Valve XVT-2869A(B)(C)	Y	Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication. (FSAR Figure 10.3-1)	RSG Isolation	25
Diesel Driven Air Compressor XAC-14	N	See Attachment 1, Item 4 for compressor attributes. (FSAR Figure 9.3-2)	RCS Cooldown	26, 27, 28
PZR PORV Block Valve XVG-8000A(C)	Y	Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication.	RCS Depressurization	29
PZR PORV (air operated) PCV-445A(B) PCV-444B	Y	See Attachment 1, Item 4 for PZR PORV attributes. Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication. (FSAR Figure 5.1-1, Sh.2)	RCS Depressurization	30, 31
High Head Injection Valves MVG-8801A(B)	Y	Valve status (open/closed) is indicated on the MCB by its associated control switch light. ESF Monitor Light on MCB provides RG-1.97 Category 2 valve position indication. (FSAR Figure 6.3-1)	SI Termination	32

Remarks for Table 16:

1. One of many indicators of a SGTR.

2. Pressurizer pressure and level decrease for primary-to-secondary leak flows in excess of normal makeup.

3. During the RCS depressurization to equalize pressure with the ruptured SG, pressurizer level is restored.

4. Prior to terminating Safety Injection (SI), the operator verifies the RCS is subcooled, an adequate secondary heat sink exists (i.e., minimum required EF flow is available or narrow range SG level is being maintained in at least one intact SG), RCS wide range pressure is stable or increasing, and pressurizer level is within the normal range.

5. Unless manually initiated by the operator, SI actuation occurs on low pressurizer pressure.

6. The RCS is depressurized until wide range RCS pressure is less than ruptured SG pressure with pressurizer level within the normal range.

7. For a large SGTR, pre-trip feed flow to the ruptured SG decreases with constant or rising level in the ruptured SG.

8. Post-trip following EF isolation, ruptured SG level continues to rise until break flow is terminated.

9. EF flow is manually controlled to ensure the SG tubes remain covered within the ruptured SG and at the post-trip control level within the intact SGs.

10. The operator determines the core exit thermocouple cooldown target (i.e., temperature that will maintain the RCS subcooled when pressure is equalized between the RCS and ruptured SG) based on steam line pressure for the ruptured SG.

11. An adequate secondary heat sink is confirmed by ensuring the minimum required EF flow is being provided and level is available in at least one SG.

12. Designed and installed per NUREG-0578, Item 2.1.3.b and NUREG-0737, Item II.F.2. The 51 thermocouple signals are routed via two separate environmentally qualified, safety related channels. Circuits terminate at safety related thermocouple isolator cabinets in a cable spreading room directly below the control room. Fifty-one isolated channels are then routed to the plant computer which is the primary display. Sixteen dedicated channels (2/train/core quadrant) are isolated and separated from the computer circuits and are routed as associated safety related to the core subcooling margin monitor, which is the backup thermocouple display and a RG-1.97 Category 1 instrument.

13. Designed and installed per NUREG-0578, Item 2.1.3.b and NUREG-0737, Item II.F.2. Provides backup core exit thermocouple display.

14. The PORV on the ruptured SG is closed, using the MCB manual/auto controllers, to isolate flow from the ruptured SG. Closed position verified using its associated MCB Control Switch light.

15. The PORVs on the intact SGs are opened, using the MCB manual/auto controllers, to cool down the RCS.

16. The SG PORVs form part of the main steam pressure boundary upstream of the main steam isolation valves (MSIVs), and thus are safety grade. The electrical and control air appurtenances for the SG PORVs are, however, non-safety grade. Each SG PORV can also be operated locally using a hand wheel. The valve operators and hand wheels are accessible from permanent platforms and are located within the Intermediate and Auxiliary Buildings at the 436-ft elevation.

17. The operator positions the EF flow control values to control SG level in the intact SGs.

18. The operator closes the EF flow control valves to isolate flow to the ruptured SG once the SG tubes are covered. Valve status (open/closed) is verified using the MCB control switch light, a RG-1.97 Category 2 indication.

19. The Reactor Protection System provides a main feedwater isolation signal to A and B trains on low Tavg coincident with reactor trip [closes feedwater isolation valves (FWIVs) only].

20. A safety injection signal causes closure of the FW isolation valves, FW control valves, and FW Bypass Control Valves and trip of the FW pumps.

21. If SG B or SG C is ruptured and at least one MDEFP is running (as indicated by the MCB RG-1.97 Category 2 motor amp indication, EF flow indications and pump status indication light), steam is isolated to the TDEFP by closing its steam supply inlet valve.

22. Not Used.

23. Part of SS pressure boundary. Valves fail closed on loss of air or control power and are automatically closed on Phase A Containment Isolation or EF pump start. Operator confirms valve is closed on the ruptured SG using position indication on MCB.

24. Operator closes main steam drain isolation valve(s) on ruptured SG from MCB to achieve isolation.

25. Part of the SS pressure boundary. Valves fail closed on loss of air or control power. Operator closes valve(s) on ruptured SG from MCB to achieve isolation.

26. Motive air is needed for the PORVs on the intact SGs to support cooldown of the RCS and for long term event recovery as discussed in Attachment 1, item 4.

27. On loss of offsite power, the normal IA compressors are lost.

28. The diesel driven air compressor is automatically started in the event of a loss of power to the in-plant compressors or loss of instrument air header pressure.

29. At least one block valve must be verified as open or opened to make a pressurizer PORV available. This is accomplished from the MCB via inspection of its associated open/closed status light indication or operation of its associated safety grade control switch.

30. The pressurizer PORVs form part of the RCS pressure boundary upstream and are thus safety grade. Control switch lights on MCB provide valve status and are RG-1.97, Category 2. The normal control function to manipulate these valves is classified as non-safety grade. Manual operation from the MCB is, however, provided by a Class IE circuit. Two of the three PORVs (PCV-444B & 445A) have a continuous back-up source of air in the form of air tanks inside containment. No operator action is required to align the back-up air supply.

31. One pressurizer PORV will be manually opened from the MCB to equalize RCS and ruptured SG pressure.

32. To terminate SI, the parallel cold leg injection valves (XVG-8801A/B) are closed from the MCB.

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ATTACHMENT 2

IMPACT ASSESSMENT FOR FUEL THERMAL CONDUCTIVITY DEGRADATION

Fuel Thermal Conductivity Degradation

Reference: Nuclear Regulatory Commission (NRC) issued Information Notice (IN) 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011.

Thermal Conductivity Degradation (TCD) can result in increased fuel temperatures during normal plant operation which would then impact the initial conditions for the SGTR analyses. Although detailed fuel performance calculations have not been performed, the potential effects of TCD on the VCSNS SGTR analyses were assessed and are discussed below.

Current Licensing Basis Analysis

The licensing basis analysis to determine the input to the SGTR doses is a conservative mass and energy balance to determine the break flow and steam releases following an SGTR and assumes break flow termination at 30 minutes. The licensing basis analysis does not model a plant-specific or fuel-specific maximum fuel temperature, but assumes a conservatively high fuel temperature, which bounds the expected effects of TCD. Thus, the licensing basis analysis is not adversely impacted.

Supplemental T/H Analysis for Mass Release Inputs to the Dose Analysis

To demonstrate that the licensing basis analysis provides conservative releases for input to the dose analyses, a more realistic "confirmatory" analysis is performed modeling operator actions and considering that break flow may extend beyond 30 minutes. For the confirmatory analysis, a high fuel temperature is conservative as it increases steam releases. The effects of TCD were evaluated and it was determined that although the calculated releases could potentially increase, the offsite and control room dose results would not change since sufficient conservatism was added to the originally calculated values (i.e., mass transfer input to the dose calculations were increased by 10 percent and 5 percent margin was added to the calculated dose). Thus, the reported results of the confirmatory calculation (Table 14 in enclosure of this LAR) and the conclusion that the licensing basis analysis is conservative remain valid.

Dose Analysis

The SGTR dose analyses do not include specific modeling of fuel temperatures and are therefore not directly impacted by TCD. The potential impact on the SGTR doses from TCD is limited to that discussed above in relation to the input to dose analyses.

Supplemental MTO Analysis

To demonstrate that overfill of the ruptured steam generator does not occur, a more realistic analysis is performed modeling operator actions and considering that break flow may extend beyond 30 minutes. For the margin-to-overfill analysis, a lower fuel temperature is conservative as it decreases steam releases, and so is not impacted by the effects of TCD.

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ATTACHMENT 3

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by the Virgil C. Summer Nuclear Station (VCSNS) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Bruce L. Thompson at (803) 931-5042.

COMMITMENT	DUE DATE
VCSNS is to revise the VCSNS FSAR to reflect the updated SGTR Analyses.	120 Days after Issuance of Amendment
VCSNS is to add the operator action times used in the license basis changes in steam generator tube rupture analysis covered in LAR-12-04269 to the operator requalification program.	120 Days after Issuance of Amendment