FIRST SUBMITTAL OF

SAFETY ANALYSIS RESULTS

TO SUPPORT REPLACEMENT STEAM GENERATOR

TECHNICAL SPECIFICATION CHANGES

FOR THE

VIRGIL C. SUMMER NUCLEAR STATION

APRIL 1993

SCE&G

SUBMITTAL SCHEDULE AND FORMAT TO SUPPORT
STEAM GENERATORS REPLACEMENT TECHNICAL SPECIFICATION CHANGES
FOR THE VIRGIL C. SUMMER NUCLEAR STATION

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LIST OF ACRONYMS & ABBREVIATIONS

ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BEF	Best Estimate Flow
BIT	Boron Injection Tank
CHG/SI	Charging/Safety Injection
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
DCD	Document Control Desk
DECL	Double-Ended Cold Leg
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
ECT	Eddy Current Testing
$F\Delta H$	Hot Channel Enthalpy Rise Factor
F_Q	Total Peaking Factor
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
GPM	Gallons per Minute
LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
LOL/TT	Loss of Load/Turbine Trip
LPZ	Low Population Zone
M/E or M&E	Mass and Energy
MMF	Minimum Measured Flow
MSLB	Main Steam Line Break
MWt	Megawatt Thermal
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
$OP\Delta T$	Overpower Delta T
$OT\Delta T$	Overtemperature Delta T
PCT	Peak Clad Temperature
PLOF	Partial Loss of Flow
RB	Reactor Building
RBCU	Reactor Building Cooling Unit
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RSG	Replacement Steam Generator

LIST OF ACRONYMS & ABBREVIATIONS

RSR	Relative Stability Ratio
RTP	Rated Thermal Power
RWST	Refueling Water Storage Tank
SCE&G	South Carolina Electric & Gas Company
SER	Safety Evaluation Report
SI	Safety Injection
SIS	Safety Injection System
SG	Steam Generator
SGR	Steam Generator Replacement
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
TA	Total Allowance
T_{AVG}	RCS Average Temperature
T_{HOT}	Vessel Outlet Temperature
T_{COLD}	Vessel Inlet Temperature
TDF	Thermal Design Flow
VANTAGE+	V+
VCSNS	Virgil C. Summer Nuclear Station

EXECUTIVE SUMMARY

This document is the first submittal of safety analysis results to the NRC in support of Replacement Steam Generator (RSG) Technical Specification changes for the Virgil C. Summer Nuclear Station (VCSNS). The Table of Contents lists all topics that will be addressed to support the RSG Technical Specification changes and reflects the schedule of all planned submittals. As indicated in the Table of Contents, the topics addressed in this partial submittal are as follows:

- Section 2.1 Design Power Capability Parameters
- Section 3.4 Containment Analyses (LOCA)
- Section 3.5 Steam Generator Tube Rupture Analysis
- Section 3.8 Radiological Analysis
- Section 3.9.8 Application of Leak-Before-Break Methodology

Several submittals are planned to support the RSG Technical Specification changes. As noted in a previous letter from SCE&G [John L. Skolds to DCD "VCSNS Proposed Schedule for Submittal of Information Supporting Steam Generator Replacement (REM 6000)", dated June 4, 1992], it is expected that early submittal of discrete packages of analyses will assist the NRC with meeting SCE&G's Fall 1994 SG Replacement (SGR) schedule.

The major submittal in support of SGR is scheduled for 10/29/93. This submittal will contain all supporting safety analyses and evaluations, the 10CFR50.92 No Significant Hazards Determination, and the proposed changes to the VCSNS Technical Specifications which support the RSGs and associated revised design power capability parameters. The final submittal is scheduled for December 1993 and will address only the Small Break Loss of Coolant Accident analysis results and associated proposed Technical Specification changes, if needed.

It should be noted that, where possible, the analyses and evaluations performed to support the RSGs incorporate the Engineered Safeguards Design Rating ("stretch" power rating) of 2900 MWt core power. This conservatively bounds the current licensed core power of 2775 MWt and is used to minimize future reanalysis for a potential stretch power application. However, it should be emphasized that approval is not being sought at this time for operation at stretch power.

A summary of the results of the analyses and evaluations contained in this spring 1993 submittal is as follows:

The analyses and evaluations performed to support the RSGs bound a range of operating conditions for VCSNS. Four cases are presented which define a range of primary operating temperatures from 572°F to 587.4°F and a range of steam generator tube plugging levels from 0% to 10%. This will provide SCE&G with the flexibility to select the appropriate primary temperatures on a cycle-by-cycle basis necessary to achieve full megawatt electric output and to adjust the temperature as necessary to compensate for steam generator tube plugging or to perform end-of-cycle T_{avg} coastdown.

The results of the LOCA containment analysis demonstrate that:

1) For the short term containment response, comprising the reactor building subcompartment analyses, the current analyses for the steam generator compartment and reactor cavity are shown to remain bounding. The surge line and spray line mass and energy releases are shown to increase; however, large margins continue to be maintained between the calculated and design

differential compartment pressures. In summary, the structural integrity of the Reactor Building subcompartments will be maintained for the RSG and associated changes in plant operating conditions.

2) For the long term analysis of the Reactor Building integrity, it was determined that the RSGs, when analyzed at conditions corresponding to the stretch power level of 2912 MWt NSSS, have a small impact on the Reactor Building pressures and temperatures following a design basis LOCA. The new calculated peak pressure remains well below the Reactor Building design pressure, resulting in a minimum design margin of 26.4%. In addition, reduced Reactor Building Cooling Unit performance was assumed in combination with the larger RSGs; the impact of this is higher Reactor Building temperatures and pressures in the long term. However, these increases can be accommodated within existing design margins with no impact on plant equipment.

The reactor core and reactor coolant iodine and noble gas fission product activities were recalculated to support the radiological consequence analyses in FSAR Chapter 15 with the RSGs, revised design power capability parameters, and transition to VANTAGE+ fuel. These fission product activities are utilized in the calculation of offsite doses presented in Section 3.8.3. Section 3.8.2 provides the specific VANTAGE+ core, coolant, and fuel handling accident source terms with a comparison to the VANTAGE 5 core and to a generic 2900 MWt core.

The Steam Generator Tube Rupture (SGTR) Analysis results are summarized in Table 3.5-2. The parameters in Table 3.5-2 are for the primary to secondary break flow and the atmospheric steam release via the faulted steam generator and are based on the VCSNS SGTR sensitivity analysis. These results can be used to determine the radiological consequences on SGTR for VCSNS with replacement steam generators when operated within the bounds of the design power capability parameters. Note that these results account for Steam Generator replacement, increasing power to the stretch power limit, hot leg temperature reduction, and Steam Generator Tube Plugging programs and are bounding for operation within the ranges of parameters listed in Table 3.5-1.

A reconciliation was performed for the recently approved Leak-Before-Break (LBB) methodology, WCAP 13206, to incorporate the effects of hardware changes and a potential stretch power application. The hardware changes include removal of several SG support snubbers, removal of crossover leg whip restraints, and the replacement of the SGs. The reconciliation of the LBB is contained in WCAP 13605 which is included as Appendix 5 of this document. The results of the calculations performed to reconcile the elimination of the RCS primary loop breaks for the VCSNS under the new loop configuration and potential stretch power application demonstrate that the conclusions reached in WCAP 13206 remain unchanged. Thus, it was concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis for VCSNS.

In summary, the safety analyses and evaluations provided with this submittal demonstrate acceptable results in each case, incorporating the revised operating conditions associated with the RSGs.

2 BASIS FOR EVALUATIONS/ANALYSES PERFORMED

2.1 DESIGN POWER CAPABILITY PARAMETERS

2.1.1 Discussion of Parameters

Design power capability parameters were developed for the Virgil C. Summer Nuclear Station (VCSNS) to encompass both the Delta-75 Replacement Steam Generators (RSGs) and the stretch power level (2912 MWt NSSS). The parameters developed are bounding for the lower power level of 2787 MWt NSSS that is the current licensed power for VCSNS (Reference 1). For licensing purposes, four parameter cases were developed in order to examine a range of operating conditions: 0% to 10% steam generator tube plugging and a vessel average temperature ranging from 572.0°F to 587.4°F. The safety analyses presented in this submittal considered the case(s) which is most conservative for the specific analysis areas. The parameter cases are provided in Table 2.1-1 and are explained in detail below.

Cases 1 and 2, calculated for 0% and 10% Steam Generator Tube Plugging (SGTP), respectively, incorporate the conservatively low Reactor Coolant System (RCS) Thermal Design Flow (TDF) (92,600 gpm/loop), as well as the current licensed T_{avg} value of 587.4°F (Reference 1). The TDF of 92,600 gpm/loop was selected such that adequate margin (approximately 8%) exists between TDF and best estimate predictions of RCS flow, assuming the Delta-75 steam generator with 10% SGTP. The RCS Best Estimate Flow (BEF) is based on Delta-75 steam generator hydraulic characteristics, reactor coolant pump performance curves, and RCS pressure drop data. Cases 1 and 2 are used for those analyses [e.g., non-LOCA, Departure from Nucleate Boiling (DNB)-related] where high RCS temperatures and low RCS flow are bounding.

Cases 3 and 4 (0% and 10% SGTP) incorporate TDF and the lowest reactor vessel T_{avg} considered, 572°F. The reduced temperature conditions allow for constant operation at reduced temperatures, or end-of-cycle T_{avg} coastdown capability. These cases are used for analyses where low vessel inlet temperature is bounding (NSSS design transients for the cold leg) or where low steam pressure is bounding (e.g., consideration of pressure drop across the steam generator tubes).

2.1.2 References

1. Virgil C. Summer Nuclear Station Final Safety Analysis Report

TABLE 2.1-1

DESIGN PERFORMANCE CAPABILITY PARAMETERS FOR VCSNS DELTA-75 REPLACEMENT STEAM GENERATORS ANALYSES

For all parameter cases:

Parameter	
NSSS Power, MWt	2912
RCP Power, MWt	12
Core Power, MWt	2900
Core Bypass Flow, %	8.9
RCS Design Pressure, psia	2250
Thermal Design Flow, gpm/loop	92,600
Minimum Measured Flow, gpm total	283,500
Best Estimate Flow, gpm/loop	102,600
Mechanical Design Flow, gpm/loop	107,100
Fuel Design	Vantage + (V+)
Fuel Peaking Factors	$F_q = 2.45$, $F_{\Delta H} = 1.62$
Positive Moderator Temp. Coef., pcm/°F	+7

	High T_{avg} Cases:		Low T_{avg} Cases:	
Parameter	Case 1	Case 2	Case 3	Case 4
Coolant Temperatures, °F				
Core Outlet	627.7	627.7	613.5	613.5
Vessel Outlet	621.9	621.9	607.4	607.4
Core Average	592.8	592.8	577.1	577.1
Vessel Average	587.4	587.4	572.0	572.0
Vessel/Core Inlet	552.9	552.9	536.6	536.6
Zero Load	557.0	557.0	557.0	557.0
Steam Generator				
Feedwater Temperature, °F	440.0	440.0	440.0	440.0
Moisture Carryover, %	0.1	0.1	0.1	0.1
Steam Temperature, °F	540.4	538.4	523.7	521.7
Steam Pressure, psia	966	950	839	824
Steam Flow, million lb/hr.	12.84	12.83	12.77	12.76
Tube Plugging, %	0	10	0	10

3.4 CONTAINMENT ANALYSES

3.4.1 LOCA Mass and Energy Releases

3.4.1.1 Long Term LOCA Mass and Energy Releases

Introduction:

Rupture of any of the piping carrying pressurized high temperature reactor coolant water, termed a Loss of Coolant Accident (LOCA), will result in release of steam and water into the containment. This, in turn, will result in an increase in the containment pressure and temperature. In the long term, the maximum overall pressure and temperature achieved within the containment are of interest. The mass and energy release rates described in this submittal form the basis of further computations to evaluate the structural integrity of the containment following a postulated LOCA.

The releases resulting from a spectrum of postulated LOCAs have been computed previously (Reference 1) for VCSNS in its present configuration. These new analyses reflect the changes due to the replacement steam generators and revised design power capability parameters, as well as reflecting improvements resulting from more realistic computer modeling.

Analytical Approach:

Description of LOCA Mass and Energy Release Transients:

The LOCA transient is typically divided into four phases:

1. Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS pressure reaches initial equalization with containment.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. For the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment and is a conservative treatment of the refill period. Thus, the refill period is conservatively assumed to begin and end at the end of blowdown.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

Tables 3.4.1-2 through 3.4.1-9 provide tabulations of the long term mass and energy release rates versus time for all four phases over the spectrum of breaks analyzed.

Selection of LOCA Break Size and Location:

Generic studies have been performed with respect to the effect of postulated break sizes on the LOCA mass and energy releases. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between reactor vessel and steam generator)
2. Cold leg (between pump and reactor vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed and described herein is the DEPS guillotine break. Pump suction break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA. In addition, the blowdown phase of a DEHL break has been investigated. The following paragraphs contain a comparison of the characteristics for each of the potential break locations, and form the rationale for this selection.

The DEHL guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generators in venting to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break.

For the hot leg break, there is no reflood peak as determined by generic studies (i.e., from the end of the blowdown period the releases would continually decrease). Therefore the reflood (and subsequent post-reflood) releases are not calculated for a hot leg break. Only the mass and energy releases for the hot leg break blowdown phase have been analyzed.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment peak pressure. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break (Reference 2). During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is also reduced. Therefore, the cold leg break analysis is not usually performed.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment. This break location has been determined to be the limiting break for typical dry containment plants. The choice of this break location as the limiting break analyzed for VCSNS is consistent with other dry containment plants for the post-blowdown phase of the event.

In summary, based on previous studies, the DEPS guillotine break has historically been considered to be the limiting break location for the post-blowdown phase of the event, by virtue of its consideration of all energy sources present in the RCS. The analyses presented support the conclusions of the double-ended pump suction (DEPS) as the limiting break case for the post-blowdown period, considering both the minimum and maximum safety injection cases. This break location provides a mechanism for the release of the available energy in the Reactor Coolant System, including both the broken and intact loop steam generators.

Basis for Mass and Energy Release Calculations:

The evaluation model used for the long term LOCA mass and energy release calculations was the March 1979 model described in Reference 2. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other dry containment plants.

For the long term mass and energy release calculations, maximum operating temperatures at 102% of the plant's stretch power capability (2958 MWt) were selected as the bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. An allowance of +5.3°F is also added to the temperatures in order to account for instrument error and deadband. Additionally, conservative primary/secondary heat transfer coefficients and conservative RCS metal heat transfer coefficients were selected to maximize the rate of energy transfer.

The initial RCS pressure in this analysis is based on a nominal value of 2250 psia with an allowance of +50 psia, which accounts for the uncertainty on pressurizer pressure. The resulting limiting pressure of 2300 psia affects the blowdown phase results only since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure. Use of a high pressure conservatively maximizes the mass and energy releases for two reasons. First, the rate at which the RCS blows down is initially more severe at the higher RCS pressure (2300 psia); and second, the RCS has a higher fluid density at 2300 psia (assuming a constant temperature) and subsequently has a higher RCS mass available for release. Thus, 2300 psia initial pressure was selected as the limiting case for the long term mass and energy release calculations.

A fuel allowance is included in the long term mass and energy calculation and subsequent LOCA containment integrity calculation to conservatively maximize the core stored energy. Fuel densification effects are included. The margin in core stored energy was chosen to be +15 percent, which is well above the calculated uncertainty in initial fuel temperature. Thus, the fuel conditions are very conservative and provide a bounding analysis for 17 x 17 V+ fuel.

The mass and energy calculations were performed for both minimum and maximum safety injection flow rates.

In summary, the following items ensure that the mass and energy releases are conservatively calculated for maximum containment pressure:

1. Maximum expected operating temperature of the reactor coolant system

2. Allowance in temperature for instrument error and dead band (+5.3°F)

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Nominal power level of 2900 MWt

5. Allowance for calorimetric error (+2% of power)

6. Conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core stored energy effect of fuel densification

8. Margin in core stored energy (+15%)

9. Allowance for RCS pressure uncertainty (+50 psi)

The initial conditions used in these analyses are summarized in Table 3.4.1-1.

Blowdown Mass and Energy Release Calculations:

The SATAN-VI code is used for computing the blowdown transient and is the same as that used for the ECCS calculation in Reference 3. The approved methodology for the use of this model is described in Reference 2.

Tables 3.4.1-2 and 3.4.1-3 present the calculated mass and energy releases for the blowdown phase of the DEPS and DEHL breaks, respectively. Break flow time histories from each side of the guillotine break are tabulated, where Break Flow Path No. 1 represents the flow from the reactor vessel outlet side of the break, and Break Flow Path No. 2 represents the flow from the reactor vessel inlet side of the break. The mass and energy release for the DEPS break and the DEHL break, given in Tables 3.4.1-2 and 3.4.1-3, terminate 19.6 and 18.2 seconds, respectively, after the initiation of the postulated accident.

Reflood Mass and Energy Release Calculations:

The WREFLOOD code is used for computing the reflood transient and is a modified version of that used in the ECCS calculation in Reference 3. The approved methodology for the use of this model is described in Reference 2.

To enhance the mass and energy evaluation model described in Reference 2, steam/water mixing in the broken loop has been included in this analysis. This enhancement is justified, as supported by test data, and its basis is summarized below:

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. Since the mass and energy of the steam released is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data (Reference 4) validates the containment integrity reflood steam/water mixing model. This data is from a 1/3 scale test, the largest scale data available; and it most closely simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, one group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests was reviewed and discussed in detail in Reference 2. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

The following justification is also noted. The limiting break for the containment integrity peak pressure analysis during the post-blowdown phase is the DEPS break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam, which is not condensed by ECC injection in the intact RCS loops, passes around the downcomer and through the broken loop cold leg and pump before venting to containment. This steam also encounters ECC injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. The condensed portion of steam is credited in the analysis. This assumption is justified based upon the postulated break location and the actual physical presence of the ECC injection nozzle. A description of the tests and test results is contained in References 2 and 4.

The reflood methodology described above and in Reference 2 has been utilized and approved by the NRC for Catawba Units 1 and 2, Indian Point 2 and 3, McGuire Units 1 and 2, Sequoyah Units 1 and 2, Millstone Unit 3, Shearon Harris, and Beaver Valley Unit 2.

Tables 3.4.1-4 and 3.4.1-5 present the calculated mass and energy release for the reflood phase of the DEPS break, with minimum and maximum safety injection respectively. Flow time histories from each side of the DEPS break are tabulated, where Break Flow Path No. 1 represents the flow through the outlet of the steam generator and Break Flow Path No. 2 represents reverse flow through the reactor coolant pump. A significantly higher mass and energy release occurs during the period the accumulators are injecting (from 22.3 to 43.8 seconds for minimum and maximum safety injection as illustrated in Tables 3.4.1-4 and 3.4.1-5). The transient results for the principal parameters during reflood are given in Tables 3.4.1-6 and 3.4.1-7 for the minimum and maximum safety injection DEPS break cases.

<u>Post-Reflood Mass and Energy Release Calculations:</u>

The FROTH code is used for computing the post-reflood transient. The methodology for the use of this model is described in Reference 2. The mass and energy release rates calculated by FROTH are provided for use in the containment analysis and are intended to apply until the time of containment depressurization.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in Reference 5, and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANSI/ANS Standard (1979).

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Tables 3.4.1-8 and 3.4.1-9 present the two phase (froth) mass and energy release data for the DEPS break minimum and maximum safety injection cases. Flow time histories from each side of the DEPS break are tabulated, where Break Flow Path No. 1 represents the flow through the outlet of the steam generator, and Break Flow Path No. 2 represents reverse flow through the reactor coolant pump.

The mass and energy release rates calculated by FROTH are used in the containment analysis until the time of recirculation. Following recirculation, credit was taken for cooling the ECCS fluid via the residual heat removal system heat exchangers. Tables 3.4.1-8 and 3.4.1-9 do not reflect this credit; see Section 3.4.3.2 for details.

<u>Mass and Energy Sources:</u>

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 3.4.1-10, 3.4.1-11, and 3.4.1-12. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 3.4.1-13, 3.4.1-14, and 3.4.1-15.

The components included in the mass and energy calculations are:

1. Reactor Coolant System water
2. Accumulator water
3. Pumped injection water
4. Decay heat
5. Core stored energy
6. Reactor Coolant System metal
7. Steam Generator metal
8. Steam Generator secondary energy
9. Secondary transfer of energy (feedwater into and steam out of the steam generator secondary)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the energy released by this reaction to be of any significance. System parameters needed to perform confirmatory analyses are provided in Table 3.4.1-1.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis and that the guidelines presented in Standard Review Plan (Reference 17) have been satisfied.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time

3. End of refill time
4. End of reflood time
5. Time of full depressurization

The methods and assumptions used to release the various energy sources are given in Reference 2 (except as noted previously) which has been approved as a valid evaluation model by the NRC.

<u>Conclusions:</u>

Long term mass and energy release rates were computed for a double-ended guillotine break located in the pump suction piping. For this postulated break, the releases were computed assuming maximum and minimum safeguards were operational. In addition, the releases in the first phase of the transient (blowdown) were computed resulting from a postulated break in the hot leg of the RCS. The results of the computations were similar to those previously calculated (Reference 1). Tables 3.4.1-2 through 3.4.1-9 summarize the results for the long term releases.

3.4.1.2 Short Term LOCA Mass and Energy Releases

Introduction:

Reactor Building subcompartment analyses are performed to demonstrate the adequacy of containment internal structures and attachments when subjected to dynamic localized pressurization effects that occur during the first few seconds following a design basis pipe break accident. Subsequent to the postulated rupture, the pressure builds up at a faster rate than the overall containment pressure, thus imposing differential pressure across the walls of the structure.

This section evaluates the effects of the RSGs and associated design power capability parameters on short term LOCA mass and energy releases used as input to Reactor Building subcompartment analyses.

Current Licensing Basis:

The current licensing basis analyses of short term LOCA mass and energy releases are presented in Section 6.2.1.3.10.2 of the FSAR. The mass and energy releases were generated with the Westinghouse 1975 M&E model (Reference 6) for the following breaks to support subsequent analyses of the reactor cavity, steam generator, and pressurizer compartments:

1. 150 in^2 Cold Leg Break (reactor cavity blowdown)
2. Double-Ended Cold Leg Break
3. Double-Ended Hot Leg Break
4. Double-Ended Pressurizer Surge Line Break
5. Pressurizer Spray Line Break

Mass and energy releases for these breaks are presented in FSAR Tables 6.2-41 through 6.4-45.

Since completion of the FSAR analyses, a Leak-Before-Break (LBB) Methodology (Reference 7) has been applied which eliminates the dynamic effects of postulated primary pipe ruptures from the design basis. Application of LBB means that the 150 in^2 cold leg break, DECL break, and DEHL breaks need not be considered in the structural design basis for the Reactor Coolant System or Reactor Building. Since the Reactor Coolant System piping has been eliminated from consideration, only the large branch nozzles must be considered for design verification for any change made to the facility.

Impact of RSGs and Changes in Plant Parameters:

The RSGs impact on short term mass and energy releases is expected to be very small. However, the revised design power capability parameters, which support vessel average temperatures ranging from 572°F to 587.4°F at full power, affect the short term mass and energy releases. The decrease in full power operating temperatures will increase the short term mass release rate, which in turn can potentially cause subcompartment conditions to be more severe. To minimize this impact during the steam generator replacement, the LBB methodology has been re-examined to incorporate the effects of hardware changes (RSGs, snubber removal, and pipe whip restraint modification) and changes in the design power capability parameters. This reconciliation of the LBB for the primary piping (Reference 8) revalidates the conclusions reached in Reference 7 and is documented in Section 3.9.8. With credit for LBB, mass and energy releases used for subcompartment design are bounding for the smaller RCS nozzle breaks (surge line, RHR line, and the accumulator nozzles) in combination with the RSGs and revised operating parameters.

Short Term M&E for the Reactor Cavity:

The break sizes associated with the surge line, RHR lines and the accumulator nozzles are less than 150 in^2. The lower mass and energy releases from the smaller RCS nozzle breaks more than offset the initial RCS conditions penalties associated with the revised operating parameters. The RHR lines, surge line, and accumulator lines are also outside the reactor vessel cavity region and will result in minimal asymmetrical pressurization in the reactor cavity region. Therefore, considering the break size reduction with LBB and the proximity of the small breaks relative to the reactor vessel cavity, the current mass and energy release for the 150 in^2 break (VCSNS's original licensing basis) can be used to bound the effects of the RSGs and revised operating parameters.

Short Term M&E for the SG Compartments:

The break sizes associated with the surge line, RHR lines, and the accumulator nozzles are significantly less than the double-ended hot and cold leg breaks utilized in the original SG subcompartment analysis. The lower mass and energy releases from the smaller RCS nozzle breaks more than offset the initial RCS conditions penalties. Therefore, the current M&E releases for the RCS loop breaks (VCSNS's original licensing basis) can be used to bound the effects of the RSGs and revised operating parameters.

Short Term M&E for the Pressurizer Compartment:

Surge and spray line breaks are postulated in the pressurizer and surge tank compartments to evaluate subcompartment pressurization. No break size reduction is possible for these compartments, since the LBB methodology has only been applied to the large primary piping.

The mass and energy releases from the surge line break are used to evaluate the pressurizer compartment. These releases are affected by the initial temperature conditions of the fluid since they are linked directly to the critical mass flux which increases with decreasing temperatures. Since the revised design power capability parameters will allow RCS temperatures to decrease below current design values, an increase in the surge line mass and energy releases must be evaluated in support of the RSGs.

Based on the change in peak critical mass flux with decreasing temperature, the impact of the RSGs and revised design power capability parameters can be bounded by increasing the current surge line mass release rates by 15%. Although not as limiting for the pressurizer compartment, similar considerations indicate that the release rates for breaks in the spray line can be bounded by increasing the current mass release by 10%.

Conclusions:

The revised design power capability parameters associated with the RSGs will allow lower operating temperatures which will tend to increase short term mass and energy releases. To offset the impact on the SG compartments and reactor vessel cavity, Leak-Before-Break is credited to justify that the current mass and energy releases for the RCS loop breaks (plant's original licensing basis) can be used to bound the effects of the RSGs and revised operating parameters. Increases in surge line and spray line mass and energy must be considered for the pressurizer compartment to accommodate the reduced operating temperature. Evaluations conclude that the impact on the pressurizer compartment can be conservatively bounded by increasing the surge and spray line mass release by factors of 15% and 10%, respectively.

For the RSGs and revised design power capability parameters, the subcompartment design verifications should be based on the following:

1. Reactor Cavity: 150 in^2 Cold Leg Break (FSAR Table 6.2-41)

2. SG Compartments: Double-Ended Cold Leg Break (FSAR Table 6.2-42) and Double-Ended Hot Leg Break (FSAR Table 6.2-43)

3. Pressurizer Compartment: Double-Ended Pressurizer Surge Line Break (FSAR Table 6.2-44 with the mass release rate increased by 15%) and Pressurizer Spray Line Break (FSAR Table 6.2-45 with mass release rate increased by 10%).

TABLE 3.4.1-1

SYSTEM INITIAL CONDITIONS

Power Level (Includes +2% Allowance for Instrument Error and Deadband)	2958 MWt
Vessel Average Temperature (Includes +5.3°F Allowance for Instrument Error and Deadband)	592.7°F
Core Inlet Temperature (Includes 5.3°F Allowance for Instrument Error and Deadband)	558.2°F
Mass of Reactor Coolant	421.3×10^3 lbm
Reactor Coolant Pressure (Includes +50 psi Allowance for Instrument Error and Deadband)	2300 psia
Initial Steam Generator Steam Pressure	966 psia
Assumed Maximum Containment Back Pressure	71.7 psia
ECCS Accumulator	
- Temperature	120°F
- Pressure	628 psia
RWST Temperature	95°F

TABLE 3.4.1-2

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION GUILLOTINE

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.101	41226.3	22798.3	22509.4	12407.1
0.201	41151.8	22857.9	24598.8	13572.4
0.600	42021.0	24036.7	21813.2	12091.2
0.901	41596.5	24453.0	20262.5	11265.6
1.40	38613.1	23653.5	19519.5	10841.8
2.00	33824.8	21885.0	19227.8	10682.1
2.50	26966.8	18578.2	17874.7	9954.5
2.70	20342.8	14290.5	17235.4	9609.8
2.90	17530.7	12472.6	16664.7	9302.6
3.20	14967.0	10742.3	15945.2	8913.6
4.20	11851.4	8639.7	14257.3	7960.9
4.80	11211.8	8445.6	13362.6	7447.3
5.20	8680.8	7660.3	13609.6	7581.3
5.40	7950.9	7142.5	13426.4	7477.7
6.20	7694.5	6610.6	12890.6	7182.4
7.20	8353.3	6306.3	12178.7	6784.8
8.00	8031.9	6080.8	11699.2	6511.9
9.80	6443.7	5402.5	10478.1	5829.6
11.6	5529.9	4696.0	9229.6	5146.6
14.2	4260.9	3693.6	7415.4	4158.4
14.6	4128.9	3611.2	7237.4	3944.6
14.8	4059.6	3585.0	8616.4	4612.5
15.0	3961.4	3556.0	6301.6	3367.6
15.2	3886.7	3551.5	11251.0	5853.2
15.4	3704.1	3494.3	11196.1	5851.6
15.6	3677.5	3611.6	4829.9	2511.8
15.8	34[illegible]7.4	3525.7	9022.3	4356.0
16.0	3270.5	3501.4	8927.5	4374.1
16.2	3090.6	3493.0	4650.6	2292.0
16.6	2467.6	3023.5	8768.5	3941.9
16.8	2123.2	2624.3	5556.8	2540.9
17.2	1618.6	2015.9	3758.2	1743.9
18.4	571.0	720.5	1256.4	759.7
19.6	0.0	0.0	0.0	0.0

TABLE 3.4.1-3

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE-ENDED HOT LEG GUILLOTINE

TIME SECONDS	BREAK PATH NO. 1 FLOW LBM/SEC	BREAK PATH NO. 1 FLOW THOUSAND BTU/SEC	BREAK PATH NO. 2 FLOW LBM/SEC	BREAK PATH NO. 2 FLOW THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.100	40707.9	26964.9	28641.3	18613.5
0.200	37093.4	24611.9	23845.4	15400.7
0.300	35362.1	23347.9	21758.7	13867.2
0.601	34739.7	22929.6	19114.6	11631.3
1.20	31342.0	21403.5	17380.0	10059.9
2.20	23965.5	17302.9	17132.6	9580.2
2.80	21069.8	15272.0	17270.0	9571.5
3.30	19800.2	14176.8	16976.1	9390.0
4.00	19119.2	13341.5	15537.5	8628.8
4.40	19570.4	13333.2	14463.4	8077.3
5.00	21265.5	13802.9	12433.7	7023.0
5.20	14974.3	11011.2	11792.4	6691.5
6.00	16239.9	11358.0	9807.0	5662.3
6.20	17613.8	11934.2	9465.8	5482.3
6.60	26967.4	17319.2	8901.8	5180.9
7.80	26423.9	16211.8	7318.3	4332.4
8.60	24989.3	15306.5	6187.5	3768.5
8.80	15237.9	9142.2	5918.2	3639.9
9.20	15813.8	9546.9	5406.1	3402.4
9.40	10036.9	7348.0	5187.3	3306.9
9.60	9692.6	7209.3	4991.7	3222.3
10.4	11042.5	7754.2	4447.3	2969.2
10.8	13961.6	9556.2	4243.7	2864.6
11.4	12073.9	8363.3	3891.7	2698.3
11.8	5710.9	4926.3	3559.5	2557.1
12.4	4768.5	4341.9	2870.3	2325.5
13.2	2690.8	3036.4	1845.3	1999.9
13.8	1959.9	2320.3	1118.0	1385.5
15.6	882.5	1101.1	565.1	711.9
16.2	921.1	961.9	320.5	407.1
16.4	329.8	405.2	412.8	524.3
17.0	1043.3	749.3	188.9	240.7
17.4	0.0	0.0	119.2	152.8
18.2	0.0	0.0	0.0	0.0

TABLE 3.4.1-4

REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MIN SI

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
19.6	0.0	0.0	0.0	0.0
20.3	0.0	0.0	0.0	0.0
20.4	68.0	80.4	0.0	0.0
20.6	64.8	76.6	0.0	0.0
22.1	137.6	162.6	0.0	0.0
22.2	148.9	176.0	514.1	56.1
22.3	369.6	438.4	3994.2	446.3
22.4	537.9	640.0	5659.3	660.0
22.5	597.1	711.3	6181.8	745.3
22.8	641.4	764.5	6577.3	804.3
23.7	625.9	746.1	6420.0	808.9
25.7	570.8	679.8	5943.3	755.2
26.7	545.9	649.8	5718.8	730.4
27.7	523.0	622.3	5508.6	707.2
29.7	482.6	573.9	5127.6	665.5
31.7	448.1	532.5	4791.0	628.7
33.7	418.2	496.7	4490.3	596.1
35.7	392.0	465.3	4218.8	566.7
36.8	285.1	337.8	2990.3	441.1
37.0	283.5	335.9	2971.2	439.0
38.8	270.2	320.0	2807.5	421.2
39.8	285.1	337.8	3019.6	431.3
41.8	272.1	322.3	2856.9	414.0
43.8	260.7	308.8	2716.4	398.5
44.8	265.5	314.4	255.2	151.0
51.8	239.5	283.5	244.2	135.5
61.8	209.6	248.0	231.6	118.0
76.8	176.3	208.5	217.9	99.0
92.8	152.6	180.4	208.3	86.1
138.8	126.7	149.7	197.6	72.1
174.8	125.2	148.0	196.6	70.7
190.8	127.2	150.4	199.1	71.8
198.8	128.8	152.2	206.1	73.8
214.8	129.4	152.9	225.4	77.7
231.5	125.2	147.9	247.2	80.7

TABLE 3.4.1-5

REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAX SI

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
19.6	0.0	0.0	0.0	0.0
20.3	0.0	0.0	0.0	0.0
20.4	68.0	80.4	0.0	0.0
20.6	64.8	76.6	0.0	0.0
22.1	137.6	162.6	0.0	0.0
22.2	148.9	176.0	514.1	56.1
22.3	369.6	438.4	3994.2	446.3
22.4	537.9	640.0	5659.3	660.0
22.5	597.1	711.3	6181.8	745.3
22.8	641.4	764.5	6577.3	804.3
23.7	625.9	746.1	6420.0	808.9
25.7	570.8	679.8	5943.3	755.2
26.7	545.9	649.8	5718.8	730.4
27.7	523.0	622.3	5508.6	707.2
29.7	482.6	573.9	5127.6	665.5
31.7	448.1	532.5	4791.0	628.7
33.7	418.2	496.7	4490.3	596.1
35.7	392.0	465.3	4218.8	566.7
36.8	285.1	337.8	2990.3	441.1
37.0	283.5	335.9	2971.2	439.0
38.8	270.2	320.0	2807.5	421.2
39.8	314.9	373.2	3412.5	454.9
40.8	307.4	364.4	3308.0	447.2
41.8	301.3	357.0	3236.5	439.2
43.8	289.7	343.3	3101.5	424.2
44.8	151.5	179.1	733.7	152.2
46.8	150.7	178.2	735.2	151.8
62.8	145.0	171.3	746.0	149.0
64.8	144.3	170.5	747.3	148.7
80.8	138.9	164.2	757.6	146.1
102.8	132.0	156.0	771.5	142.9
162.8	120.2	142.1	794.3	139.6
172.8	118.4	139.9	797.8	138.9
238.8	106.9	126.4	820.2	133.7
239.5	106.8	126.2	820.5	133.6

TABLE 3.4.1-6

PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MIN SI

TIME SEC	FLOODING TEMP DEG F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWN HEIGHT FT	FLOW FRACT	INJECTION (LBM/SEC) TOTAL	ACCU	SPILL	ENTHALPY BTU/LBM
19.6	269.3	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
20.2	264.3	31.931	0.000	0.63	3.07	0.000	12601.6	12601.6	0.0	89.57
20.3	261.2	40.513	0.000	1.08	3.44	0.000	12386.3	12386.3	0.0	89.57
21.2	258.5	3.371	0.338	1.52	8.95	0.397	11417.8	11417.8	0.0	89.57
22.1	257.6	3.134	0.471	1.66	14.33	0.423	10709.1	10709.1	0.0	89.57
22.7	256.4	7.122	0.553	1.79	15.59	0.719	8259.1	8259.1	0.0	89.57
23.7	254.0	6.137	0.645	2.00	15.60	0.708	7555.4	7555.4	0.0	89.57
24.7	252.0	5.573	0.684	2.16	15.60	0.706	7190.6	7190.5	0.0	89.57
27.7	247.6	4.686	0.728	2.52	15.60	0.693	6330.7	6330.7	0.0	89.57
33.7	242.4	3.801	0.748	3.07	15.60	0.666	5132.1	5132.1	0.0	89.57
38.8	240.4	2.783	0.748	3.41	15.60	0.588	3237.7	3237.7	0.0	89.57
39.8	240.2	2.876	0.750	3.47	15.60	0.600	3469.3	3070.8	0.0	86.52
40.8	240.0	2.825	0.750	3.53	15.60	0.595	3371.5	2971.8	0.0	86.42
43.8	239.5	2.697	0.750	3.70	15.60	0.585	3131.0	2725.6	0.0	86.13
44.8	239.4	2.731	0.751	3.76	15.49	0.587	403.6	0.0	0.0	63.01
49.8	239.5	2.564	0.750	4.03	14.83	0.582	408.5	0.0	0.0	63.01
59.8	241.8	2.286	0.748	4.54	13.82	0.570	416.3	0.0	0.0	63.01
70.8	246.7	2.051	0.746	5.04	13.10	0.558	422.3	0.0	0.0	63.01
82.8	253.4	1.857	0.746	5.54	12.68	0.544	426.8	0.0	0.0	63.01
96.8	261.1	1.697	0.747	6.06	12.53	0.531	430.1	0.0	0.0	63.01
110.8	267.4	1.594	0.749	6.55	12.62	0.521	431.6	0.0	0.0	63.01
124.8	272.7	1.529	0.751	7.00	12.87	0.514	432.4	0.0	0.0	63.01
142.8	278.5	1.483	0.756	7.56	13.32	0.511	433.0	0.0	0.0	63.01
158.8	282.9	1.462	0.761	8.03	13.80	0.510	433.2	0.0	0.0	63.01
176.8	287.1	1.451	0.768	8.54	14.36	0.511	433.2	0.0	0.0	63.01
182.8	288.4	1.450	0.770	8.71	14.56	0.511	433.2	0.0	0.0	63.01
194.8	290.8	1.457	0.775	9.04	14.93	0.515	433.0	0.0	0.0	63.01
198.8	291.5	1.458	0.776	9.15	15.03	0.517	432.9	0.0	0.0	63.01
212.8	293.9	1.445	0.782	9.53	15.32	0.521	432.7	0.0	0.0	63.01
231.5	296.8	1.388	0.788	10.00	15.52	0.520	433.1	0.0	0.0	63.01

TABLE 3.4.1-7

PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MAX SI

TIME SEC	FLOODING TEMP DEG F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWN HEIGHT FT	FLOW FRACT	INJECTION (LBM/SEC) TOTAL	INJECTION (LBM/SEC) ACCU	INJECTION (LBM/SEC) SPILL	ENTHALPY BTU/LBM
19.6	269.3	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
20.2	264.3	31.931	0.000	0.63	3.07	0.000	12601.6	12601.6	0.0	89.57
20.3	261.2	40.513	0.000	1.08	3.44	0.000	12386.3	12386.3	0.0	89.57
21.2	258.5	3.371	0.338	1.52	8.85	0.397	11417.8	11417.8	0.0	89.57
22.1	257.6	3.134	0.471	1.66	14.33	0.423	10709.1	10709.1	0.0	89.57
22.7	256.4	7.122	0.553	1.79	15.59	0.719	8259.1	8259.1	0.0	89.57
23.7	254.0	6.137	0.645	2.00	15.60	0.708	7555.4	7555.4	0.0	89.57
24.7	252.0	5.573	0.684	2.16	15.60	0.706	7190.5	7190.5	0.0	89.57
27.7	247.6	4.686	0.728	2.52	15.60	0.693	6330.7	6330.7	0.0	89.57
33.7	242.4	3.801	0.748	3.07	15.60	0.666	5132.1	5132.1	0.0	89.57
38.8	240.4	2.783	0.748	3.41	15.60	0.588	3237.7	3237.7	0.0	89.57
39.8	240.2	3.069	0.752	3.47	15.60	0.624	3902.0	2979.5	0.0	83.29
40.8	239.9	3.018	0.751	3.53	15.60	0.615	3788.5	2866.7	0.0	83.11
44.8	239.3	2.003	0.739	3.77	15.60	0.465	979.5	0.0	0.0	63.01
50.8	239.9	1.970	0.740	4.02	15.60	0.465	979.6	0.0	0.0	63.01
62.8	243.2	1.910	0.742	4.53	15.60	0.464	979.7	0.0	0.0	63.01
74.8	248.4	1.852	0.744	5.01	15.60	0.464	979.8	0.0	0.0	63.01
88.8	255.8	1.786	0.747	5.55	15.60	0.464	979.9	0.0	0.0	63.01
02.8	263.1	1.721	0.751	6.06	15.60	0.464	980.0	0.0	0.0	63.01
16.8	269.2	1.670	0.754	6.55	15.60	0.465	980.0	0.0	0.0	63.01
30.8	274.4	1.628	0.758	7.02	15.60	0.467	979.9	0.0	0.0	63.01
46.8	279.5	1.566	0.762	7.53	15.60	0.469	979.9	0.0	0.0	63.01
162.8	283.8	1.512	0.766	8.01	15.60	0.471	979.8	0.0	0.0	63.01
180.8	288.0	1.453	0.771	8.53	15.60	0.474	979.8	0.0	0.0	63.01
198.8	291.5	1.395	0.776	9.01	15.60	0.477	979.7	0.0	0.0	63.01
218.8	294.8	1.333	0.781	9.52	15.60	0.480	979.7	0.0	0.0	63.01
239.5	297.7	1.270	0.788	10.00	15.60	0.484	979.6	0.0	0.0	63.01

TABLE 3.4.1-8

POST-REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MIN SI

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
231.6	118.5	149.3	319.4	91.8
266.6	116.2	146.4	321.7	91.4
401.6	106.0	133.6	331.8	90.4
406.6	107.2	135.1	330.7	89.9
446.6	105.5	133.0	332.4	89.2
451.6	106.6	134.4	331.3	88.8
486.6	105.2	132.5	332.7	88.2
491.6	106.2	133.9	331.6	87.8
526.6	104.7	132.0	333.1	87.2
531.6	105.8	133.3	332.1	86.7
561.6	104.5	131.7	333.4	86.2
566.6	105.5	133.0	332.4	85.8
596.6	104.1	131.2	333.7	85.2
601.6	105.2	132.5	332.7	84.8
636.6	103.8	130.9	334.0	87.8
641.6	104.9	132.1	333.0	87.3
676.6	103.5	130.4	334.4	86.5
746.6	104.1	131.1	333.8	84.0
806.6	102.5	129.2	335.4	85.7
836.6	103.4	130.3	334.5	84.3
861.6	102.2	128.8	335.7	83.6
911.6	102.9	129.6	335.0	81.4
931.6	101.8	128.3	336.1	84.1
971.6	102.4	129.1	335.5	82.1
1106.6	100.9	127.2	337.0	81.6
1436.6	101.0	127.3	336.9	80.7
1436.7	72.6	90.0	365.3	85.8
1579.0	72.6	90.0	365.3	85.8
1579.1	70.0	80.5	367.9	32.0
2393.9	70.0	80.5	367.9	32.0
2394.0	74.0	85.0	397.1	73.3
3599.9	74.0	85.0	397.1	73.3
3600.0	57.9	66.6	413.2	76.3
3600.1	43.3	49.8	427.8	63.3
10000.0	31.5	36.2	439.6	65.1
100000.0	16.8	19.4	454.2	67.2
1000000.0	7.2	8.3	463.9	68.7

TABLE 3.4.1-9

POST-REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAX SI

	BREAK PATH NO. 1 FLOW		BREAK PATH NO. 2 FLOW	
TIME SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
239.6	118.3	149.5	864.8	128.7
259.6	117.0	147.9	866.1	128.5
264.6	118.1	149.3	864.9	128.0
289.6	116.5	147.2	866.6	127.7
294.6	117.6	148.6	865.5	127.3
314.6	116.3	146.9	866.8	127.0
344.6	117.1	148.0	866.0	121.9
369.6	115.4	145.8	867.7	121.7
374.6	116.4	147.2	866.6	121.3
429.6	114.7	145.0	868.4	124.1
434.6	115.9	146.4	867.2	123.6
464.6	114.4	144.6	868.6	123.1
499.6	115.4	145.9	867.6	121.8
529.6	113.9	144.0	869.1	121.2
534.6	115.0	145.3	868.1	120.8
589.6	113.4	143.3	869.7	119.5
624.6	114.2	144.3	868.9	118.1
649.6	113.0	142.8	870.1	117.6
704.6	113.8	143.8	869.3	119.1
749.6	112.4	142.0	870.7	117.9
774.6	113.1	143.0	869.9	116.7
809.6	112.0	141.6	871.1	115.7
829.6	112.9	142.6	870.2	114.7
889.6	111.5	141.0	871.5	115.9
914.6	112.3	141.9	870.7	114.6
984.6	111.1	140.4	872.0	114.8
1349.6	111.2	140.6	871.8	113.5
1349.7	73.2	91.1	1012.7	220.1
1569.3	72.1	89.8	1013.8	220.4
1569.4	69.6	80.0	1016.4	165.1
3600.0	57.4	66.0	1028.5	167.3
3600.1	43.3	49.8	1042.6	154.3
10000.0	31.5	36.2	1054.4	156.1
100000.0	16.8	19.4	1069.1	158.2
1000000.0	7.2	8.3	1078.7	159.7

TABLE 3.4.1-10

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI

	TIME (SECONDS)	0.00	19.60	19.60	231.53	1441.60	1578.99	3600.00
					MASS (THOUSAND LBM)			
INITIAL MASS	IN RCS AND ACC	609.42	609.42	609.42	609.42	609.42	609.42	609.42
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	82.26	612.11	676.83	1628.90
	TOTAL ADDED	0.00	0.00	0.00	82.26	612.11	676.83	1628.90
*** TOTAL AVAILABLE ***		609.42	609.42	609.42	691.68	1221.54	1286.26	2238.32
DISTRIBUTION	REACTOR COOLANT	421.29	47.69	54.59	105.15	105.15	105.15	105.15
	ACCUMULATOR	188.13	146.19	139.29	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	609.42	193.88	193.88	105.15	105.15	105.15	105.15
EFFLUENT	BREAK FLOW	0.00	415.53	415.53	586.53	1116.38	1181.10	2133.17
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	415.53	415.53	586.53	1116.38	1181.10	2133.17
*** TOTAL ACCOUNTABLE ***		609.42	609.42	609.42	691.68	1221.53	1286.25	2238.32

TABLE 3.4.1-11

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MAX SI

	TIME (SECONDS)	0.00	19.60	19.60	239.50	1354.60	1569.33	3600.00
					MASS (THOUSAND LBM)			
INITIAL MASS	IN RCS AND ACC	609.42	609.42	609.42	609.42	609.42	609.42	609.42
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	195.63	1292.77	1525.96	3731.14
	TOTAL ADDED	0.00	0.00	0.00	195.63	1292.77	1525.96	3731.14
*** TOTAL AVAILABLE ***		609.42	609.42	609.42	805.05	1902.20	2135.38	4340.56
DISTRIBUTION	REACTOR COOLANT	421.29	47.69	54.59	105.82	105.82	105.82	105.82
	ACCUMULATOR	188.13	146.19	139.29	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	609.42	193.88	193.88	105.82	105.82	105.82	105.82
EFFLUENT	BREAK FLOW	0.00	415.53	415.53	699.23	1796.37	2029.56	4234.74
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	415.53	415.53	699.23	1796.37	2029.56	4234.74
*** TOTAL ACCOUNTABLE ***		609.42	609.42	609.42	805.05	1902.19	2135.38	4340.56

TABLE 3.4.1-12

MASS BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	TIME (SECONDS)	0.00	18.20	18.20
		MASS (THOUSAND LBM)		
INITIAL MASS	IN RCS AND ACC	609.42	609.42	609.42
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00
	TOTAL ADDED	0.00	0.00	0.00
*** TOTAL AVAILABLE ***		609.42	609.42	609.42
DISTRIBUTION	REACTOR COOLANT	421.29	95.75	102.64
	ACCUMULATOR	188.13	116.21	109.31
	TOTAL CONTENTS	609.42	211.96	211.96
EFFLUENT	BREAK FLOW	0.00	397.46	397.46
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	397.46	397.46
*** TOTAL ACCOUNTABLE ***		609.42	609.42	609.42

TABLE 3.4.1-13

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI

	TIME (SECONDS)	0.00	19.60	19.60	231.53	1441.60	1578.99	3600.00
				ENERGY (MILLION BTU)				
INITIAL ENERGY	IN RCS,ACC,S GEN	720.39	720.39	720.39	720.39	720.39	720.39	720.39
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	5.18	38.57	48.15	189.05
	DECAY HEAT	0.00	5.54	5.54	26.56	107.39	115.06	212.59
	HEAT FROM SECONDARY	0.00	-2.33	-2.33	-2.33	1.67	1.67	1.67
	TOTAL ADDED	0.00	3.21	3.21	29.41	147.62	164.88	403.31
*** TOTAL AVAILABLE ***		720.39	723.60	723.60	749.80	868.01	885.26	1123.69
DISTRIBUTION	REACTOR COOLANT	251.33	13.54	14.16	29.77	29.77	29.77	29.77
	ACCUMULATOR	16.85	13.09	12.48	0.00	0.00	0.00	0.00
	CORE STORED	21.91	10.73	10.73	4.10	3.95	3.86	2.71
	PRIMARY METAL	126.07	119.58	119.58	98.57	60.50	57.80	41.62
	SECONDARY METAL	80.76	79.64	79.64	73.31	45.45	42.89	31.31
	STEAM GENERATOR	223.47	224.89	224.89	203.60	123.41	116.70	85.56
	TOTAL CONTENTS	720.39	461.48	461.48	409.35	263.07	251.03	190.97
EFFLUENT	BREAK FLOW	0.00	262.11	262.11	334.25	598.74	628.04	926.53
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	262.11	262.11	334.25	598.74	628.04	926.53
*** TOTAL ACCOUNTABLE ***		720.39	723.59	723.59	743.60	861.81	879.07	1117.50

TABLE 3.4.1-14

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MAX SI

	TIME (SECONDS)	0.00	19.60	19.60	239.50	1354.60	1569.33	3600.00
				ENERGY (MILLION BTU)				
INITIAL ENERGY	IN RCS,ACC,S GEN	720.39	720.39	720.39	720.39	720.39	720.39	720.39
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	12.33	82.38	116.89	443.26
	DECAY HEAT	0.00	5.54	5.54	27.24	102.41	114.53	212.59
	HEAT FROM SECONDARY	0.00	-2.33	-2.33	-2.33	2.99	3.37	3.37
	TOTAL ADDED	0.00	3.21	3.21	37.23	187.78	234.79	659.22
*** TOTA. AVAILABLE ***		720.39	723.60	723.60	757.62	908.16	955.18	1379.61
DISTRIBUTION	REACTOR COOLANT	251.33	13.54	14.16	29.98	29.98	29.98	29.98
	ACCUMULATOR	16.85	13.09	12.48	0.00	0.00	0.00	0.00
	CORE STORED	21.91	10.73	10.73	4.10	3.95	3.82	2.71
	PRIMARY METAL	126.07	119.58	119.58	97.37	61.29	57.13	41.28
	SECONDARY METAL	80.76	79.64	79.64	73.43	46.34	42.36	30.92
	STEAM GENERATOR	223.47	224.89	224.89	203.88	127.25	116.94	86.16
	TOTAL CONTENTS	720.39	461.48	461.48	408.75	268.82	250.25	191.05
EFFLUENT	BREAK FLOW	0.00	262.11	262.11	342.67	633.15	698.74	1182.37
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	262.11	262.11	342.67	633.15	698.74	1182.37
*** TOTAL ACCOUNTABLE ***		720.39	723.59	723.59	751.42	901.97	948.98	1373.41

TABLE 3.4.1-15

ENERGY BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	TIME (SECONDS)	0.00	18.20	18.20
		ENERGY (MILLION BTU)		
INITIAL ENERGY	IN RCS,ACC,S GEN	720.39	720.39	720.39
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00
	DECAY HEAT	0.00	5.87	5.87
	HEAT FROM SECONDARY	0.00	-3.48	-3.48
	TOTAL ADDED	0.00	2.39	2.39
*** TOTAL AVAILABLE ***		720.39	722.77	722.77
DISTRIBUTION	REACTOR COOLANT	251.33	23.93	24.55
	ACCUMULATOR	16.85	10.41	9.79
	CORE STORED	21.91	8.56	8.56
	PRIMARY METAL	126.07	118.22	118.22
	SECONDARY METAL	80.76	78.42	78.42
	STEAM GENERATOR	223.47	220.64	220.64
	TOTAL CONTENTS	720.39	460.18	460.18
EFFLUENT	BREAK FLOW	0.00	262.58	262.58
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	262.58	262.58
*** TOTAL ACCOUNTABLE ***		720.39	722.76	722.76

3.4.2 Short Term Containment Analysis-LOCA Reactor Building Subcompartment Analysis

Introduction:

Reactor Building subcompartment analyses are performed to demonstrate the adequacy of containment internal structures and attachments when subjected to dynamic localized pressurization effects that occur during the first few seconds following a design basis pipe break accident. Subsequent to the postulated rupture, the pressure builds up at a faster rate than the overall containment pressure, thus imposing differential pressure across the walls of the structure.

This section evaluates the effects of the Replacement Steam Generators and associated changes in design power capability parameters on the Reactor Building subcompartment analyses using the mass and energy releases presented in Section 3.4.1.2.

Current Licensing Basis:

The current licensing basis analyses for Reactor Building subcompartment pressurization are presented in Section 6.2.1.3.9 of the FSAR. The Reactor Building subcompartments (steam generator compartments, pressurizer compartment, and reactor cavity) are analyzed with respect to the appropriate design basis break to determine the peak pressure differential for the pipe rupture. Each compartment was broken down into volumes and flow paths. The steam generator and reactor cavity compartments were analyzed using the RELAP4/MOD5 code (Reference 18) whereas the transient pressure behavior of the pressurizer compartment was calculated using the FLASH-2 code (Reference 10). The subcompartments were analyzed for the largest breaks possible in each compartment:

1. Pressurizer Compartment: spray line and surge line breaks
2. SG Compartments: double-ended hot leg and cold leg breaks
3. Reactor Cavity: 150 in^2 cold leg break

These analyses are described in detail in Sections 6.2.1.3.9.1 - 6.2.1.3.9.3 of the FSAR. The results demonstrate that the overall pressure response in each compartment is well below the design differential pressure.

Impact of the RSG and Changes in Plant Parameters:

The Reactor Building Subcompartment analyses can be affected in two respects due to the RSGs and changes in the plant's operating parameters. First, the hardware modifications (i.e., RSG, feedwater pipe rerouting, removal of SG supports, etc.) can change the subcompartment flow/vent areas and, secondly, the change in RCS operating conditions can impact the mass and energy releases. These potential effects have been evaluated for each subcompartment and the conclusions are summarized below:

Reactor Cavity

As discussed in Section 3.4.1.2, the current mass and energy release for the 150 in^2 break (plant's original licensing basis) can be used to bound the effects of the RSGs and revised operating parameters. There are also no geometric changes being made within the reactor cavity as a result of the RSGs which would require a modification to the reactor cavity model described in Section 6.2.1.3.9.3 of the FSAR. Given no change in the mass and energy releases nor changes in the layout/arrangement of the cavity, it is concluded that the pressures, forces, and moments used in the original cavity design remain bounding for the RSGs and revised operating parameters.

Steam Generator Compartments

As discussed in Section 3.4.1.2, the current mass and energy releases for the double-ended hot and cold leg breaks (plant's original licensing basis) can be used to bound the effects of the RSGs and revised operating parameters. Continued use of these mass and energy releases is very conservative because of the elimination of the large double-ended breaks in the RCS piping through application of Leak-Before-Break Methodology.

The geometry of the SG compartments will be impacted by several hardware changes, including:

1. Replacing the Model D3s (preheat SGs) with Delta-75 SGs (feedring SGs).

2. Relocating the main feedwater piping to interface with the main feedwater inlet nozzle located on the upper shell of the RSGs.

3. Angularly relocating the emergency feedwater and the narrow range water level taps on the upper shell.

4. Rerouting of instrumentation tubing to accommodate increased RSG level measurement spans.

5. Removal of three of the five hydraulic snubbers on each RSG upper lateral support (Reference 15).

6. Potential removal (may be abandoned in place) of the crossover piping whip restraints from each of the three reactor coolant pipe loops (Reference 15).

A review of these modifications indicates that they will have a negligible or favorable impact on the subcompartment model for the following reasons:

1. The external and support interface dimensions of the replacement steam generators are essentially identical to those of the current Model D3 as described in WCAP-13480 (Reference 16).

2. Removal of snubbers and pipe whip restraints will increase the subcompartment vent areas and minimize the pressurization transient.

3. Piping and tubing which is being rerouted is not a significant restriction to flow.

In addition, the lower mass and energy releases from the smaller RCS nozzle breaks (per Section 3.4.1.2) will offset not only the initial RCS conditions penalties associated with the revised operating parameters but also small changes in the SG subcompartment geometry.

Therefore, with large conservatisms in mass and energy releases and no significant changes in the layout/arrangement of the SG compartment, it is concluded that the pressures, forces, and moments used in the original SG compartment design remain bounding for the RSGs and revised operating parameters.

Pressurizer Compartment

The pressurizer and surge tank compartments have been reevaluated for a 15% increase in spray line mass release rate and a 10% increase in the surge line mass release rate. The analysis was performed using the models presented in FSAR Section 6.2.1.3.9.1 and the COMPARE (Reference 9) computer code. The model definition is provided in FSAR Figures 6.2-18 and 6.2-19. All parameters are as defined in these figures with the exception of the relief area between the surge tank compartment and containment. The relief area is adjusted to reflect the plant's as-built condition (156.2 ft^2 versus 134.2 ft^2).

Table 3.4.2-1 presents the results of the analysis. The calculated differential pressure increased for the pressurizer compartment and decreased for the surge tank compartment. However, for both the surge and spray line break, large margins continue to be maintained between the calculated and design pressures. Overall, the results demonstrate that structural integrity of the pressurizer and surge tank compartments is maintained for SG replacement and changes in plant operating conditions.

<u>Conclusions:</u>

Current LOCA pressures, forces, and moments used in the original SG compartment and reactor cavity design analyses remain bounding for the RSGs and revised operating parameters. Differential pressures resulting from potential increases in surge line and spray line mass and energy releases are shown to increase in the pressurizer compartment and decrease in the surge tank compartment. However, large margins continue to be maintained between the calculated and design pressures. Based on the results of the LOCA calculations and evaluations described above, it is concluded that the structural integrity of the Reactor Building subcompartments will be maintained for SG replacement and associated changes in plant operating conditions.

TABLE 3.4.2-1

PRESSURIZER AND SURGE TANK COMPARTMENTS
COMPARISON OF RSG PRESSURES

Compartment Description	Calculated Δ Pressure (psid)	Calculated Max. Pressure (psia)	Design Δ Press (psid)	Margin %
Pressurizer RSG-Uprate/FSAR	26.0/21.9	42.9/37.8	41.4	59.2/89.0
Surge Tank RSG-Uprate/FSAR	30.6/36.7	45.0/52.0	51.3	67.6/39.8

3.4.3 Long Term Containment Analysis

3.4.3.2 LOCA Reactor Building Integrity Analysis

Introduction:

Analyses have been completed to determine the Reactor Building pressure and temperature response during postulated LOCAs using mass and energy release which incorporates the Replacement Steam Generators (RSGs) and revised design power capability parameters for the VCSNS. The results of these analyses demonstrate that the RSGs have a small impact on LOCA consequences within the Reactor Building and that the Reactor Building design conditions remain bounding.

Analytical Approach:

Method of Analysis

The Reactor Building pressure and temperature response is calculated using the CONTEMPT-LT26 (Reference 11) computer code. This is a deviation from the current LOCA licensing basis analyses (FSAR Section 6.2) which used CONTEMPT-LT22 (Reference 12). Calculations for the same mass and energy releases and modeling assumptions, however, show that both CONTEMPT-LT codes predict similar results for LOCA conditions. Thus, use of CONTEMPT-LT26 does not represent a significant change from the current LOCA licensing basis analyses. The general methodology used in the CONTEMPT-LT26 analysis is as described in FSAR Section 6.2.1.3.3 and summarized below.

Reactor Building Initial Conditions

The Reactor Building initial conditions used in the peak pressure LOCA Analysis are presented in Table 3.4.3-1. These initial conditions are consistent with those assumed in the current licensing basis analyses. The use of maximum initial Reactor Building pressure (1.5 psig), maximum initial operating temperature (120°F), and nominal relative humidity (30%) results in a conservatively high prediction of Reactor Building pressure and temperature.

Reactor Building Design and Evaluation Parameters

The Reactor Building design and evaluation parameters are presented in Table 3.4.3-2. These parameters are consistent with those assumed in the current licensing basis analyses (FSAR Table 6.2-1) except for the Reactor Building Cooling Unit performance which is discussed further below.

Reactor Building Heat Sinks

The Reactor Building structures (i.e., interior walls, exterior walls, metal components and structures, etc.) are included in the CONTEMPT-LT26 model as heat sinks. Modeling is performed in accordance with Section 6.2.1.3.4.1 of the FSAR. The passive heat sinks and associated thermal properties are identical to those used in the current licensing basis analyses and are presented in Tables 6.2-8 and 6.2-9 of the FSAR. The model incorporates the use of less than actual heat sinks, low thermal conductivities, and low volumetric heat capacities to conservatively maximize the Reactor Building pressure during the LOCA.

Condensing Heat Transfer

Condensing heat transfer is considered within CONTEMPT-LT26 for both steel and concrete surfaces. During blowdown, heat transfer to steel surfaces from the Reactor Building vapor regions is based upon the Tagami (Reference 13) correlation. Its use in CONTEMPT-LT26 is described in Section 6.2.1.3.4.1 of the FSAR. After blowdown, a condensing film heat transfer coefficient based on the Uchida correlation (Reference 14) is applied to the steel surfaces. In the long term, use of the Uchida correlation results in a conservative prediction of Reactor Building pressure and temperature during the LOCA. For concrete surfaces, the heat transfer from the Reactor Building vapor regions is based upon the Uchida correlation throughout the LOCA transient.

Reactor Building Sprays

VCSNS has two independent, 100% capacity Reactor Building spray trains which are actuated upon a high containment pressure of 12.31 psia. The assumed performance characteristics of the Reactor Building sprays are shown in Table 3.4.3-2. Consistent with the current licensing basis analyses assumptions (Section 6.2.1.3.4.2 of the FSAR), the sprays are assumed to be 100% efficient in removing heat from the Reactor Building and are conservatively assumed to be initiated within 52 seconds.

Reactor Building Cooling Units

The Reactor Building Cooling Units (RBCUs) are modeled in CONTEMPT-LT26 as described in FSAR section 6.2.1.3.4.3. It is conservatively assumed that the RBCUs start at 86.5 seconds (time that the fan coolers are operating at full design capability) following accident initiation. The RBCU heat removal rate has been conservatively reduced by more than 50% below current licensing bases analysis assumptions (FSAR Figure 6.2-15) to allow for future potential degradation in those units.

Sump Water Recirculation

During the post accident long term cooling period, safety injection pumps supply recirculated water from the Reactor Building recirculation sumps to the reactor vessel. The Residual Heat Removal (RHR) heat exchangers are placed in operation during the recirculation phase to remove energy directly from the reactor vessel to the outside environment.

The RHR heat exchangers are of the shell and U-tube type. Physical and thermal characteristics of each heat exchanger are specified in Table 3.4.3-2 as are the coolant inlet temperatures and flow rates. Heat removed by the RHR heat exchanger provides a means to cool the Reactor Building sump in the long term and is modeled within CONTEMPT-LT26 consistent with the current licensing basis analyses assumptions.

<u>Scope of Analysis:</u>

Reactor Building pressure and temperature analyses have been performed for the following:

1. DEHL guillotine break during the blowdown period.

2. DEPS break with minimum SI, minimum Reactor Building spray, and minimum RBCU performance resulting from a postulated loss of offsite power with a failure of one emergency diesel.

3. DEPS break with maximum SI, minimum Reactor Building spray, maximum RBCU performance assuming offsite power is available and a failure of one train of Reactor Building spray.

The long term Reactor Building performance (to 10^6 seconds) has also been evaluated for the DEPS break with minimum SI to verify the ability of the ECCS and Reactor Building heat removal systems to maintain the Reactor Building pressure and temperature below design limits. The DEPS break is used since this postulated break results in greater mass and energy release during the post-blowdown period as discussed in Section 3.4.1.1.

The LOCA mass and energy release data used for these Reactor Building pressure and temperature calculations are given in Section 3.4.1.1.

Reactor Building Pressure and Temperature Results:

Table 3.4.3-3 presents the peak calculated Reactor Building pressures and temperatures for the LOCAs analyzed. The current licensing basis results for the same breaks are also presented for comparative purposes. These results show the following:

1. Peak Reactor Building pressures occur near the end of blowdown for all breaks analyzed.

2. The peak Reactor Building pressures show a slight decrease from the current licensing basis analyses for the DEPS breaks.

3. The peak Reactor Building pressure occurs for the DEHL break. The peak pressure value of 45.1 psig is approximately 2.1 psig higher than the peak LOCA value from the current licensing basis analyses.

4. The peak pressure calculated for LOCA is well below the Reactor Building design pressure of 57 psig. The minimum resulting design margin is 26.4%.

Figures 3.4.3-1 through 3.4.3-6 provide the Reactor Building pressure and temperature profiles for the LOCAs analyzed.

Figures 3.4.3-1 and 3.4.3-2 for the DEPS break with minimum SI and minimum Reactor Building cooling also demonstrate the continued long term cooling capability of the Reactor Building. These profiles are conservatively generated assuming the Reactor Building spray system is shut down after 24 hours. The accident chronology for this design basis LOCA is given in Table 3.4.3-4. In the long term, the Reactor Building pressure and temperature are higher than those from the current licensing bases analyses primarily due to the assumed reduction in RBCU performance. This increase in long term Reactor Building consequences is, however, well within existing margins to design limits and can be accommodated with no detrimental impact on plant equipment.

Conclusions:

The Reactor Building pressure and temperature analyses demonstrate that the RSGs, when analyzed at conditions corresponding to the stretch power level of 2912 MWt NSSS, have a small impact on the Reactor Building pressures and temperatures following a design basis LOCA. The calculated peak pressure of 45.1 psig for the DEHL break is well below the Reactor Building design pressure of 57 psig and results in a minimum design margin of 26.4%.

Reduced RBCU performance in combination with the larger RSGs result in higher Reactor Building temperatures and pressures in the long term. These increases can, however, be accommodated within existing design margins with no impact on plant equipment.

TABLE 3.4.3-1

INITIAL CONDITIONS USED IN REACTOR BUILDING PEAK PRESSURE ANALYSIS

Reactor Building	
Pressure, psig	1.5
Temperature, °F	120
Relative Humidity, %	30
Service Water Temperature, °F	105
Refueling Water Temperature, °F	95
Stored Water	
Refueling Water Storage Tank, gal	404,000

TABLE 3.4.3-2

GENERAL CONTAINMENT DESIGN AND EVALUATION PARAMETERS

General Design Information		
Maximum Internal Design Pressure, psig		57
Maximum External Design Pressure, psig		3.5 psig
Design Temperature, °F		283
Free Volume, ft^3		1.84×10^6
Design Leak Rate, max. allowable, %/day		0.2
Engineered Safety Features	Full Capacity	Value Used for Analysis
Passive Safety Injection:		
Number of Accumulators	3	2 or 3
Pressure Setpoint, psig	600	600
Active Safety Injection:		
Residual Heat Removal Flow Rate, lb/sec	900.4	460.2 or 990.4
Reactor Building Spray System:		
Number of Lines	2	1 or 2
Number of Pumps	2	1 or 2
Number of Headers	6	3 or 6
Design Flow, gpm	5,000	2,500 or 5,000
Reactor Building Cooling Units:		
Number	4	1 or 2
Air Side Flow Rate, cfm/unit	60,270 (1)	54,200 (1)
Heat Removal Rate at 283°F, Btu/hr/unit	125×10^6	50×10^6

Notes:

(1) This parameter is used only in the Chapter 15 Radiological Consequence Analysis for particulate iodine removal post-LOCA. This parameter is not used in the Chapter 6 Pressure/Temperature Analyses.

TABLE 3.4.3-3

COMPARISON OF REACTOR BUILDING PRESSURIZATION RESULTS - LOCA

Description	Primary System Postulated Pipe Break		
Break Location	DEPS	DEPS	DEHL
Safety Injection	Min	Max	NA
RB Spray	Min	Min	NA
RB Fan Coolers	Min	Max	NA
Peak Pressure (psig) RSG Uprate/FSAR	43.7/44.7	43.7/44.3	45.1/43
Time to Peak Pressure (sec) RSG Uprate/FSAR	18/280	18/350	15/12.1
Peak Temperature (°F) RSG Uprate/FSAR	265.4/266.7	265.4/266.1	267.4/264.3

Key:

RB Design Pressure = 57 psig
RB Design Temperature = 283°F
DEPS = Double-Ended Pump Suction
DEHL = Double-Ended Hot Leg

TABLE 3.4.3-4

CHRONOLOGY OF EVENTS FOR LOCA (DEPS, MINIMUM SAFETY INJECTION)

	Time (Seconds)
1. Break Occurs	0
2. Peak RB Pressure of 43.7 psig is Reached	18.0
3. Primary System Blowdown Complete	19.6
4. Reactor Building Spray Begins	52.0
5. Reactor Building Fan Coolers Actuated	86.5
6. Recirculation Begins	3470
7. Spray System Operation Terminated	8.64E+04
8. End of Analysis	1.00E+06

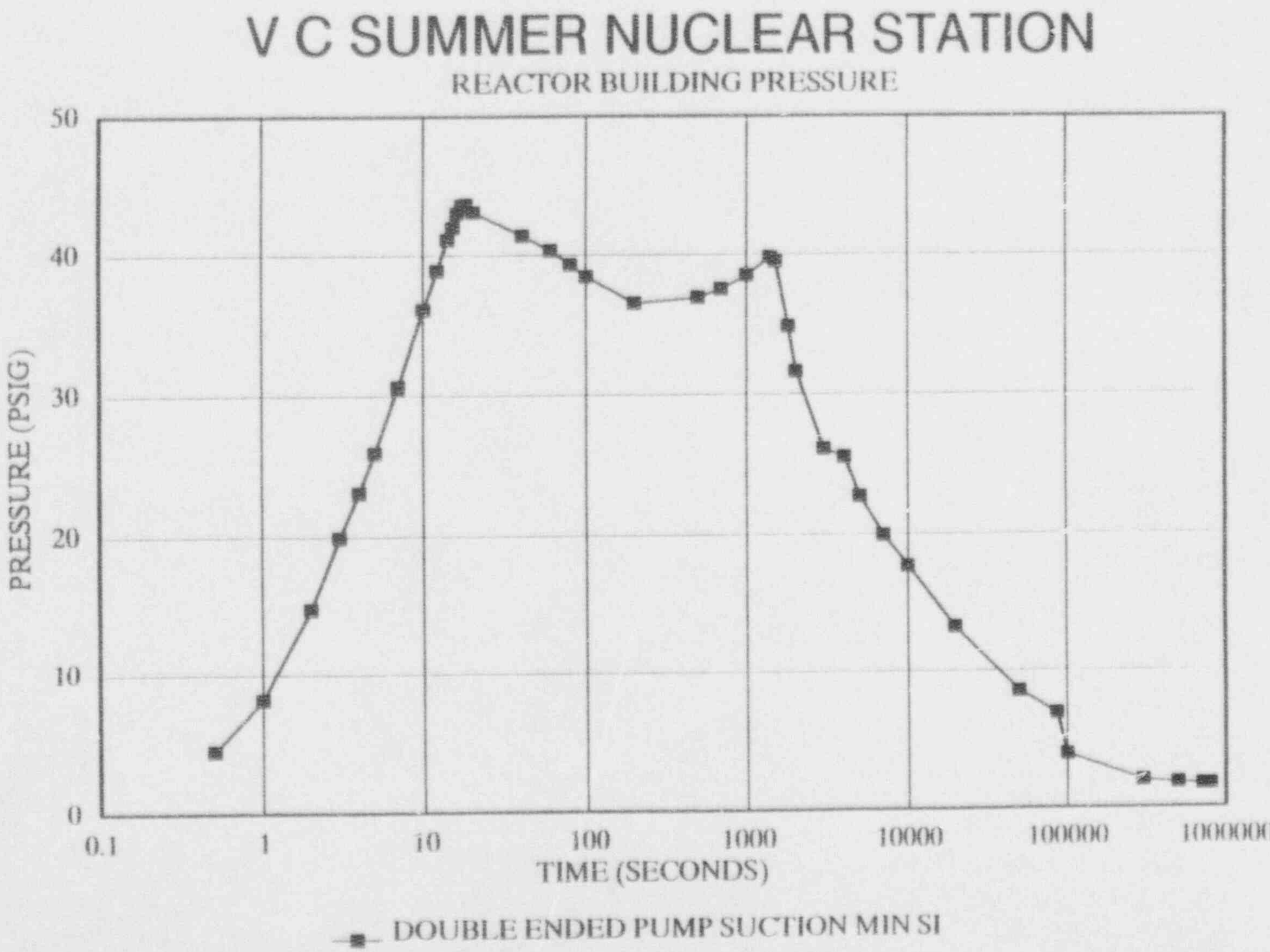
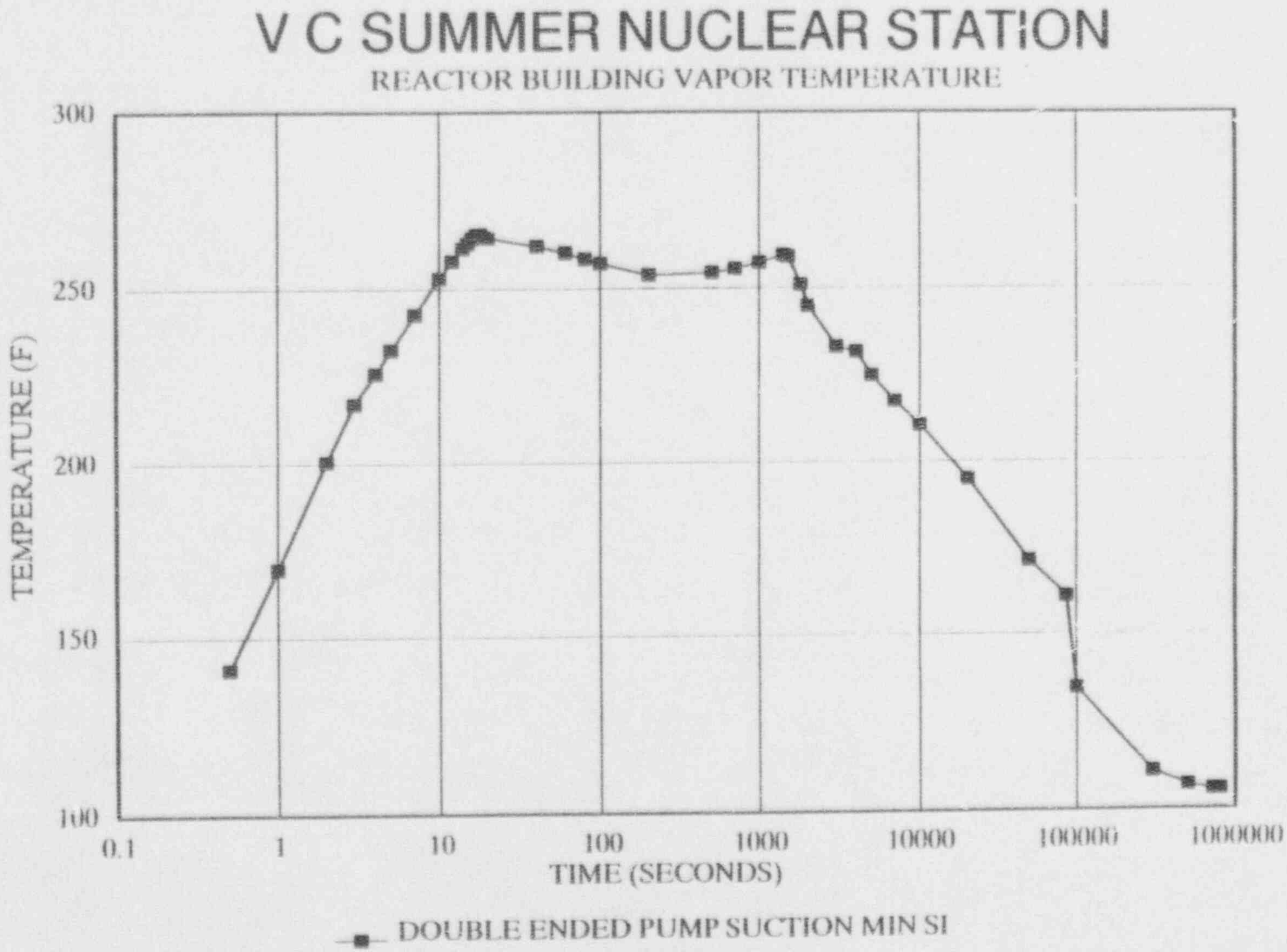

V C SUMMER NUCLEAR STATION
REACTOR BUILDING PRESSURE
PRESSURE (PSIG)
50
40
30
20
10
0
0.1
1
10
100
1000
10000
100000
1000000
TIME (SECONDS)
DOUBLE ENDED PUMP SUCTION MIN SI

FIGURE 3.4.3-1



V C SUMMER NUCLEAR STATION
REACTOR BUILDING VAPOR TEMPERATURE
300
250
200
150
100
TEMPERATURE (F)
0.1
1
10
100
1000
10000
100000
1000000
TIME (SECONDS)
DOUBLE ENDED PUMP SUCTION MIN SI

FIGURE 3.4.3-2

V C SUMMER NUCLEAR STATION

REACTOR BUILDING PRESSURE

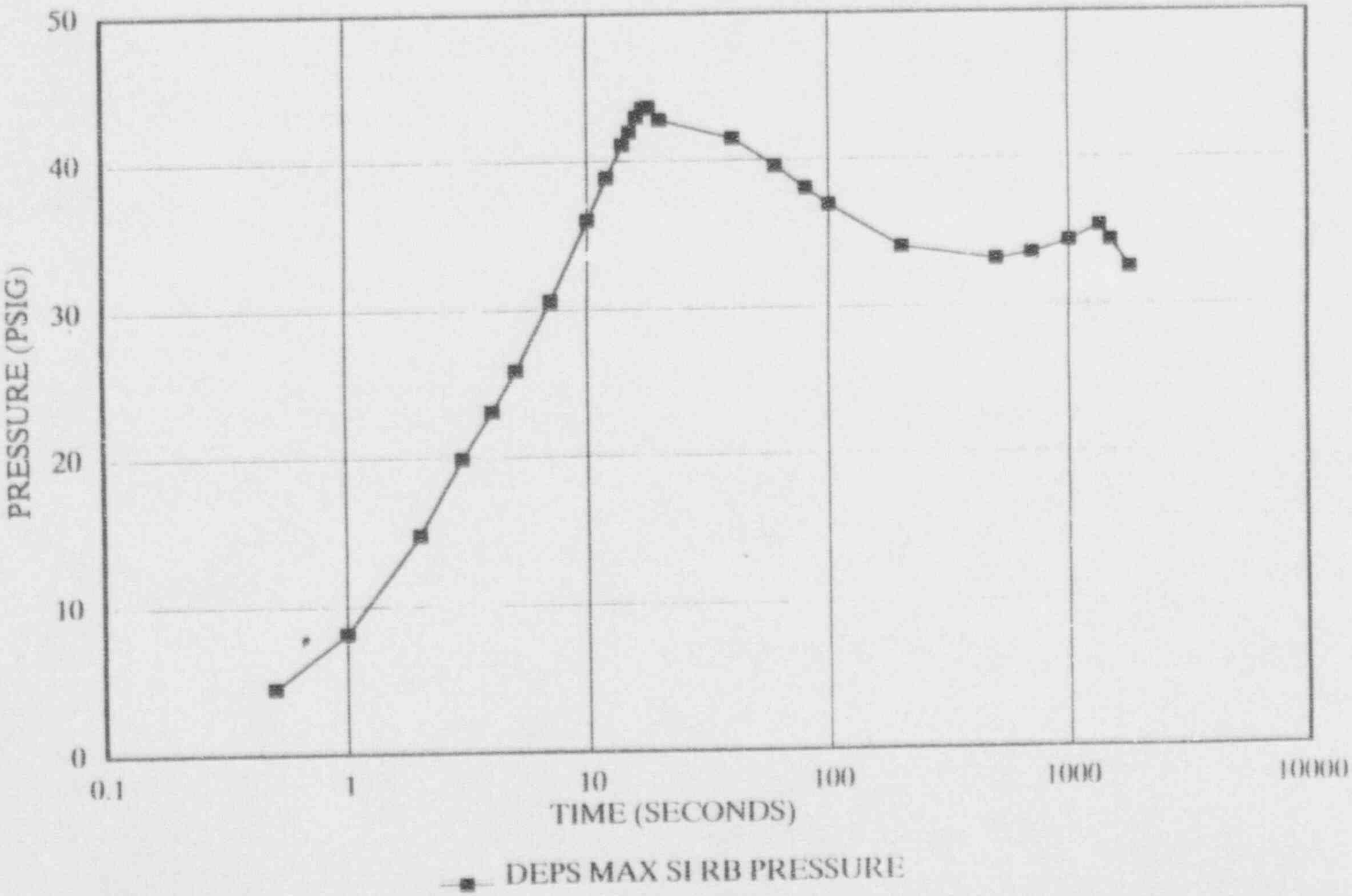
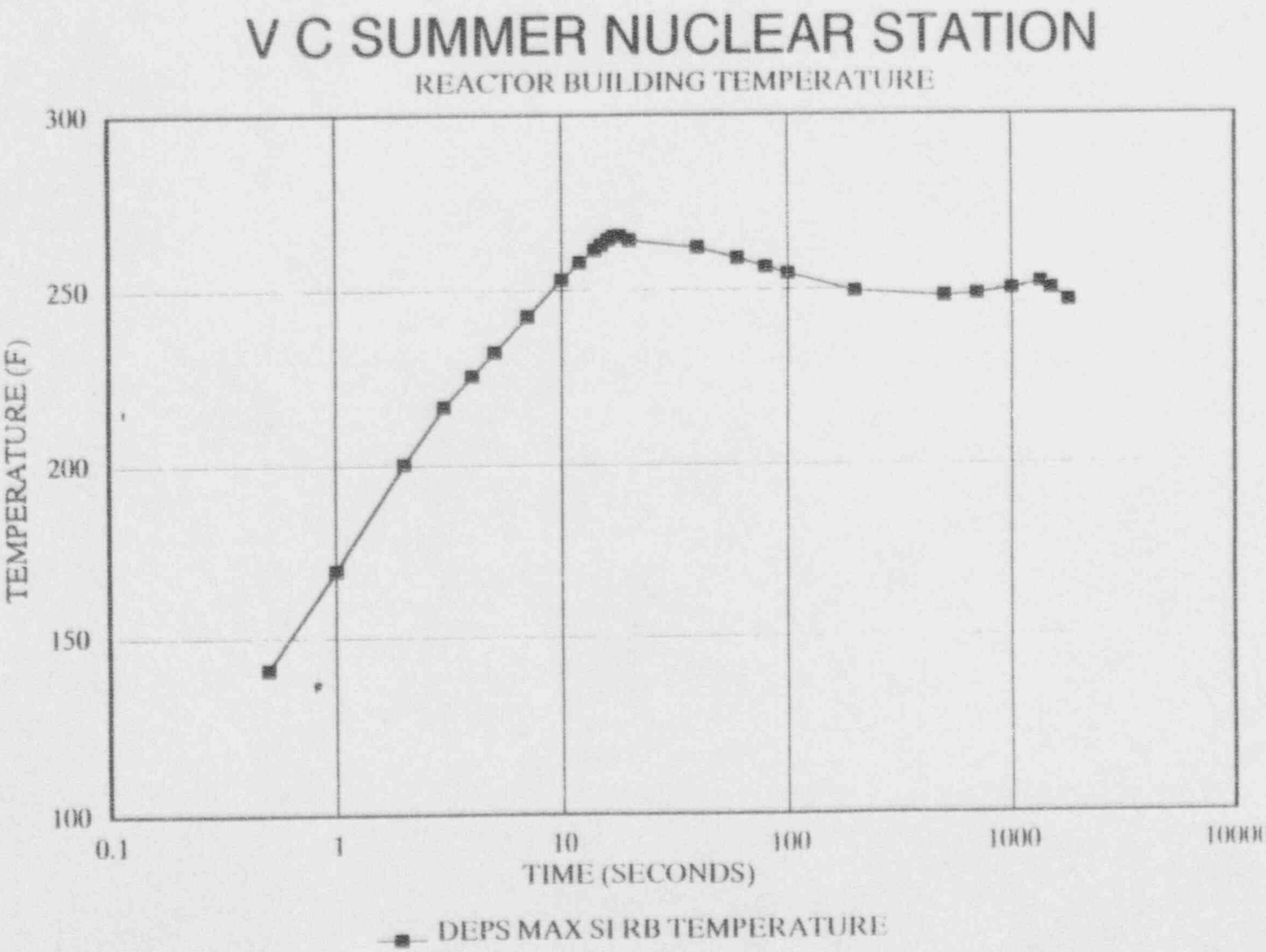


FIGURE 3.4.3-3



V C SUMMER NUCLEAR STATION
REACTOR BUILDING TEMPERATURE
300
250
200
150
100
TEMPERATURE (F)
0.1
1
10
100
1000
10000
TIME (SECONDS)
DEPS MAX SI RB TEMPERATURE

FIGURE 3.4.3-4

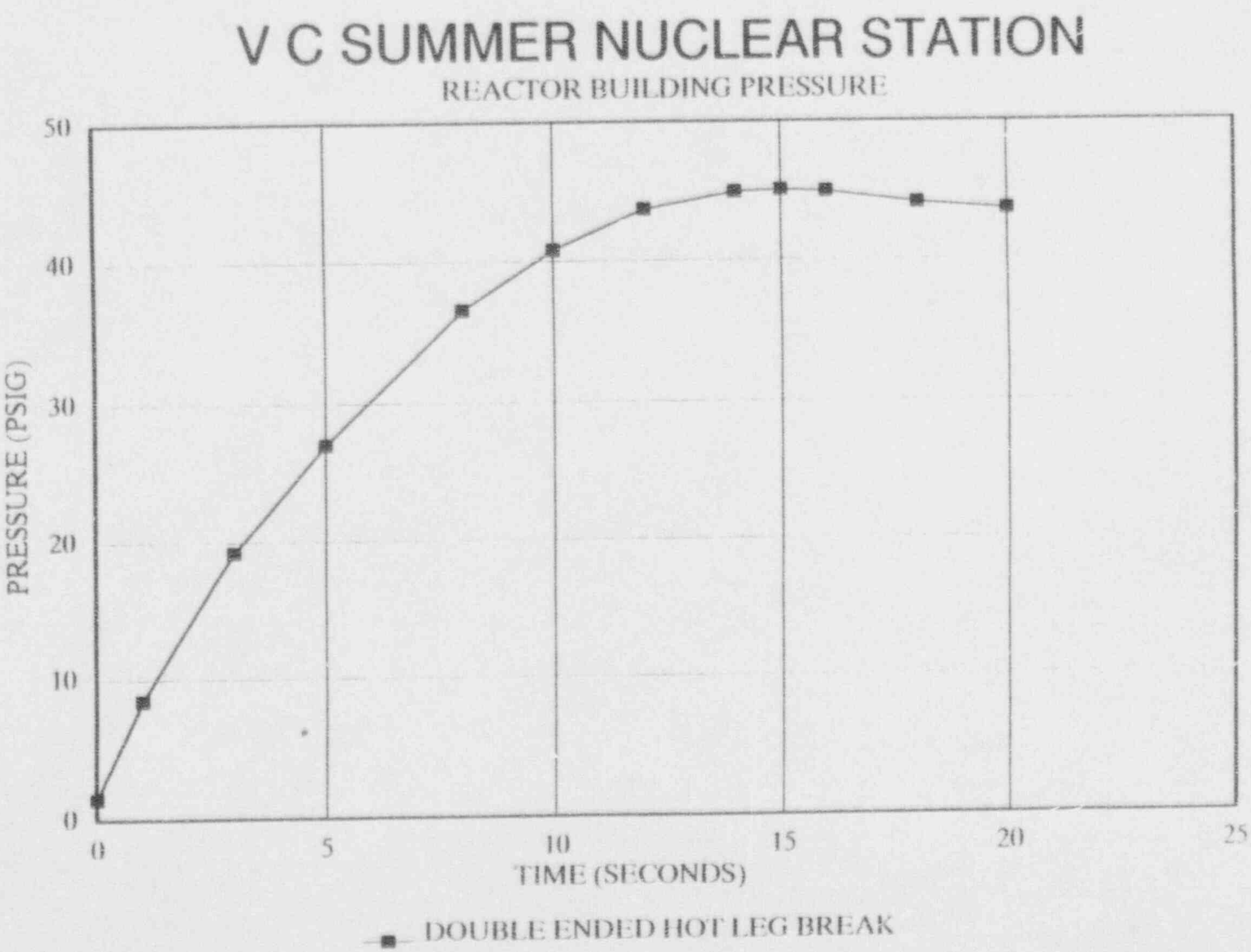

V C SUMMER NUCLEAR STATION
REACTOR BUILDING PRESSURE
PRESSURE (PSIG)
0
10
20
30
40
50
0
5
10
15
20
25
TIME (SECONDS)
DOUBLE ENDED HOT LEG BREAK

FIGURE 3.4.3-5

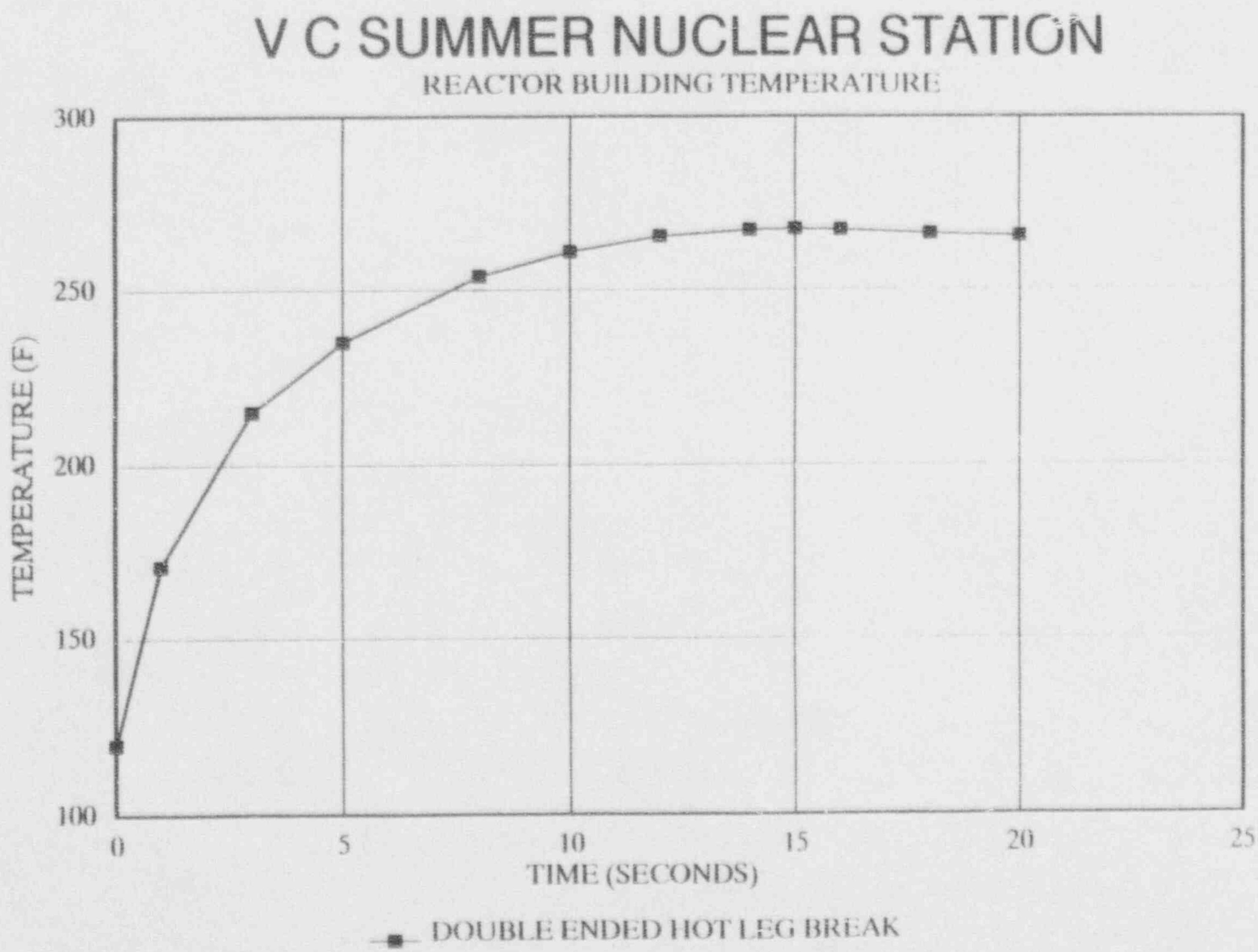

V C SUMMER NUCLEAR STATION
REACTOR BUILDING TEMPERATURE
TEMPERATURE (F)
300
250
200
150
100
0
5
10
15
20
25
TIME (SECONDS)
DOUBLE ENDED HOT LEG BREAK

FIGURE 3.4.3-6

3.4.4 References

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2. WCAP-10325-A, "Westinghouse LOCA Mass and Energy Release Model For Containment Design - March 1979 Version", April 1979.

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15. SCE&G Letter, John L. Skolds To The USNRC, "Piping Analyses For The Delta 75 Steam Generators (REM 6000-6)", RC-93-0068, March 12, 1993.

16. WCAP-13480, "Westinghouse Delta 75 Steam Generator Design and Fabrication Information For The Virgil C. Summer Nuclear Station", August 1992.

17. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, June 1987.

18. Idaho National Engineering Laboratory, "RELAP4/MOD5 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems - User's Manual", ANCR-1335, September, 1976.

3.5 STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS

Introduction:

An evaluation has been performed to determine the effect of SGR and the revised design power capability parameters on the FSAR SGTR accident analysis. A series of analyses using the methodology and assumptions that are consistent with those used for the FSAR SGTR analysis were performed to assess the impact of parameter changes associated with the steam generator replacement, increasing power to the stretch power limit, hot leg temperature reduction, and SGTP. The range of parameter changes were analyzed simultaneously to bound the operating conditions for VCSNS.

The SGTR accident analysis was performed to evaluate the radiological consequences of an SGTR accident. A complete break of a single steam generator tube was assumed. Since the RCS pressure is greater than the steam generator shell side pressure, radioactive reactor coolant is discharged into the secondary system. Also, due to the assumed coincident loss of offsite power and subsequent unavailability of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety valves. The major factors that affect the resultant offsite doses are the amount of fuel defects (level of reactor coolant contamination), the primary to secondary mass transfer through the ruptured tube, and the steam released from the faulted steam generator to the atmosphere.

The major thermal and hydraulic analysis factors which impact the offsite radiation doses calculated for the FSAR SGTR analysis are the primary to secondary break flow and the steam released from the faulted steam generator to the atmosphere. Therefore, sensitivity SGTR analyses were performed to assess the impact of steam generator replacement, increasing power to the stretch power limit, hot leg temperature reduction, and SGTP on the primary to secondary break flow and the steam released to the atmosphere via the faulted steam generator. The relevant parameters for the SGTR analysis are listed in Table 3.5-1 along with the ranges of values which are enveloped by the sensitivity analyses.

Methodology:

The SGTR accident analyzed is a double-ended rupture of a single steam generator tube. The subsequent reactor coolant loss via the ruptured tube leads to RCS depressurization and a decrease in the pressurizer level. Reactor trip and safety injection are assumed to occur simultaneously at the low pressurizer pressure safety injection signal. A loss of offsite power is assumed to occur at the time of reactor trip.

After safety injection actuation, the maximum flow from the safety injection system is assumed to be injected into the RCS until 30 minutes after the accident initiation, at which time it is assumed that the operator has completed the necessary actions to terminate the break flow.

For the determination of primary to secondary break flow prior to reactor trip and safety injection actuation, the RCS depressurization rate due to the primary to secondary break flow was calculated. The average break flow during the time period from tube rupture initiation to reactor trip was used to calculate an integrated break flow for this period. After reactor trip, the break flow rate is assumed to equilibrate at the pressure where the safety injection flow rate is balanced by the outgoing break flow rate. This resultant equilibrium break flow rate is assumed to persist until safety injection is terminated at 30 minutes. These integrated break flow values are then summed to yield the total primary to secondary break flow for the 30 minutes.

Since offsite power is assumed to be lost coincident with reactor trip, the condenser steam dump system would not be operable. Following reactor trip, the steam generator pressure increases rapidly due to the automatic turbine trip (at reactor trip) and lack of normal steam dump via the condenser. Therefore, steam is relieved through the steam generator relief valves to dissipate the plant residual heat and the core decay heat. For the SGTR analysis, it was assumed that the steam is relieved via the safety valves and the steam generators are maintained at the lowest safety valve setpoint. A mass and energy balance for the primary and secondary system was utilized to calculate the steam released via the safety valve on the faulted and intact steam generators for the 30-minute time period considered.

For the remainder of the transient, after faulted steam generator isolation, steam releases and feedwater flows for the non-faulted steam generators were obtained by an energy balance between the primary and secondary systems. For this calculation, the plant is assumed to be in a stable, no-load condition between 30 minutes and 2 hours. Afterwards, it is assumed that the operators perform a plant cooldown using the intact steam generators until the RCS temperature equals the RHR system initiation temperature. It is assumed that the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident.

To determine the impact of the steam generator replacement, increasing the power to the stretch power limit, hot leg temperature reduction, and SGTP on the SGTR accident, several sensitivity analyses were completed. Specifically, analyses were performed for a T_{avg} range of 572.0°F to 587.0°F and a steam generator tube plugging range of 0% to 10% to determine the effect on the primary to secondary break flow and steam released via the faulted steam generator for the 30 minute time period. In addition, the SGTR sensitivity analyses also incorporated bounding CHG/SI flow rates and recently revised pressure setpoint tolerances on the main steam safety valves (Reference 1). The maximum values of the primary to secondary break flow and the steam release from the faulted steam generator were determined from the sensitivities. Thus, the results of the SGTR analysis are bounding for operation of VCSNS within the range of parameters specified in Table 3.5-1.

<u>Results:</u>

The results of the VCSNS SGTR analyses for the FSAR in support of the steam generator replacement and changes in the design power capability parameters are summarized in Table 3.5-2. For the FSAR analysis, the primary to secondary break flow totaled 125,000 lbs and the steam released via the faulted steam generator was 48,000 lbs for the 30 minute time period considered. The bounding results of the sensitivity cases analyzed showed a decrease in the primary to secondary break flow to 92,900 lbs and an increase in the steam released via the faulted steam generator to 56,800 lbs. Steam releases and feedwater flows for the two non-faulted steam generators are listed in Table 3.5-2 for 0-2 hr. and 2-8 hr. time periods. Note that these results are for Steam Generator replacement, increasing power to the stretch power limit, hot leg temperature reduction, and Steam Generator Tube Plugging programs and are bounding for operation within the ranges of parameters listed in Table 3.5-1.

<u>Summary:</u>

The primary to secondary break flow and the atmospheric steam release via the faulted steam generator are summarized in Table 3.5-2 based on the VCSNS SGTR sensitivity analysis. These results can be used to determine the radiological consequences on SGTR for VCSNS with replacement steam generators when operated within the bounds of the design power capability parameters.

Reference:

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TABLE 3.5-1 VCSNS SGTR PARAMETER SENSITIVITIES	
Initial Condition	SGTR Parameter Values
Reactor Power (MWt)	2900
RCS Pressure (psia)	2250
Vessel Average Temperature Range (°F)	572.0 - 587.4
Thermal Design Flow (gpm/loop)	92,600
SG Pressure (psia)	824.0 - 966.0
SG Tube Plugging Range (%)	0 - 10

TABLE 3.5-2 VCSNS SGTR RESULTS		
	FSAR	SG Replacement & Stretch Power
SGTR Break Flow (lbm)	125,000	92,900
Steam Release (lbm) from Defective Steam Generator	48,000	56,800
Steam Release (lbm) from 2 Unaffected Steam Generators	316,000 (0-2 hr) 835,000 (2-8 hr)	381,400 (0-2 hr) 924,900 (2-8 hr)
Feedwater Flow to 2 Unaffected Steam Generators	346,000 (0-2 hr) 883,000 (2-8 hr)	371,400 (0-2 hr) 986,400 (2-8 hr)

3.8 RADIOLOGICAL ANALYSIS

3.8.1 Introduction

To support the radiological consequence analyses in Chapter 15 of the FSAR with the installation of RSGs, revised design power capability parameters, and transition to VANTAGE+ fuel, the reactor core and reactor coolant iodine and noble gas fission product activities have been recalculated. These fission product activities are utilized in the calculation of offsite doses presented in Section 3.8.3. In Section 3.8.2, the specific VANTAGE+ core, coolant and fuel handling accident source terms are presented and also compared to those of the VANTAGE 5 core and to a generic 2900 MWt core.

3.8.2 Source Terms

Introduction:

To support the radiological consequence analyses presented in Chapter 15 of the FSAR with the installation of replacement steam generators, revised power capability parameters, and transition to VANTAGE+ fuel, the reactor core and coolant iodine and noble gas fission product activities have been recalculated. The reactor core fission product activities are utilized in the calculation of offsite doses for the accidents that are postulated to result in fuel clad failure or melting and the subsequent release of fission products to the environment. These accidents include the large-break LOCA, control rod ejection, reactor coolant pump locked rotor and the fuel handling accident. Further, the plant is assumed to have operated with a small percentage of defective fuel for sufficient time to establish equilibrium concentrations of fission products in the reactor coolant.

Reactor coolant activity is a component of the total activity release assumed for the following accidents:

- Loss of Offsite Power
- CVCS Line Break
- LOCA
- Main Steam Line Break
- SGTR
- Locked Rotor
- RCCA Ejection

The salient parameters utilized in the VANTAGE+ and VANTAGE 5 fission product calculations are shown below. These parameters bound those of the current licensing basis calculation which is based on VANTAGE 5 fuel. These source terms are intended to bound future fuel cycles, within the limits of the current power capability parameters.

	Current Licensing Basis	Proposed Licensing Basis with RSGs
Fuel Type	VANTAGE 5	VANTAGE +
Core Thermal Power, Mwt	2775	2958
Average Discharge Burnup, MWD/MTU	41,000	65,370
Cycle Length, EFPD	428	480
Enrichment (wt-% U-235)	3.6	5.0

Methodology:

Each of the analyses in FSAR Chapter 15 contains assumptions regarding the fraction of core activity released as result of the accident transient. The most limiting, with regard to the release of fission products from the core, is the large-break LOCA (Section 15.4.1.4). The LOCA is based on a TID-14844 (Reference 1) release which assumes that 50% of the core iodine and 100% of the core noble gas is released to the containment. The TID iodine release is also the source term assumed for the sump solution in the determination of offsite and control room doses due to RHR system leakage outside containment. The TID release represents a core melt. Other accidents are less severe. For example, the locked rotor (Section 15.4.4.4) and control rod ejection (Section 15.4.6.4) consider the release of 15% and 10%, respectively, of the fuel rod gap activity due to cladding failures rather than fuel melting. The assumed gap fraction is 0.1 of the total core or assembly activity. Thus, the core release for the locked rotor is 0.15 (0.1) = 0.015 or 1.5%, which is significantly less than the assumed LOCA releases of 50 and 100% for iodines and noble gases.

The fuel handling accident is also a fuel rod gap release event, but it takes place during plant shutdown rather than at power. The salient fuel handling accident source term assumptions are shown below. The FSAR includes realistic and Regulatory Guide 1.25 cases (Reference 2). Only the more conservative regulatory guide case was evaluated.

	Current Licensing Basis	Proposed Licensing Basis with RSGs
Fuel Type	VANTAGE 5	VANTAGE +
Radial Peaking Factor	1.65	1.7
Number of Damaged Assemblies	1.0 (263 rods)	1.2 (314 rods)
Decay time, hours	100	100
Gap Fraction:		
All Iodine and noble gas except I-131 and Kr-85	0.1	0.1
I-131	0.1	0.12
Kr-85	0.3	0.3

In the calculation of reactor coolant fission product activity, small cladding defects in the equivalent of 1% of the fuel rods are assumed to be present at the initial core loading and uniformly distributed throughout the core.

<u>New Source Terms:</u>

The iodine and noble gas activity assumed to be released from a VANTAGE+ core following a design basis large-break LOCA (TID-14844 release, 100% of core noble gas, 50% of core iodine) is shown in Table 3.8.2-1. The iodine and noble gas activity assumed to be released from a damaged fuel assembly following a fuel handling accident is shown in Table 3.8.2-2. The reactor coolant activity is shown in Table 3.8.2-3.

Evaluation of New Source Terms:

Core Source Term

With the transition from VANTAGE 5 to VANTAGE+ fuel, there is an associated increase in the maximum fuel burnup limit. The design burnup limit for VANTAGE+ fuel is a peak fuel pin burnup of 75,000 MWD/MTU which is an increase over the current peak fuel pin burnup of 60,000 MWD/MTU.

The extension of fuel burnup has been shown to have negligible impact on the core inventory of radioactive isotopes which are of concern (i.e., the short half-life iodines and noble gases) in evaluating the radiological consequences of the accidents described in Chapter 15 of the FSAR. Compared with fuel operated without extended burnup limits (defined by the NRC as region average discharge burnup of less than 38,000 MWD/MTU), the short-lived iodine isotopes are found to decrease slightly as a result of operation to the extended burnup level, and the short-lived noble gases, with the exception of Xe-133 and Xe-135, also show a decreased core inventory associated with extended fuel burnup. The inventory of Kr-85 increases significantly with extended burnup. Because of its long half-life, Kr-85 continues to build up in the core rather than reaching equilibrium, as do the shorter half life nuclides. Increased cycle length and U-235 enrichment are primarily responsible for the increase in Kr-85 with extended fuel burnup. Although Kr-85 has increased, this nuclide is a weak gamma emitter and the increase will contribute little to the calculated whole-body doses.

Table 3.8.2-4 shows the activity assumed to be released from the core following a large-break LOCA (TID-14844 release: 50% of core iodine, 100% of core noble gas) and the salient core parameters. This is provided for VANTAGE 5, VANTAGE+, and for a generic 2900 MWt core. The generic 2900 MWt data is included to help illustrate the concepts described above.

The results show that the inventory of all nuclides have increased in the transition from VANTAGE 5 to VANTAGE+. However, the increases are primarily due to analyzing at the increased core power (approximately 6.6%). Comparison of VANTAGE+ with the generic data shows identical I-134 and 135 values. Since VANTAGE+ is based on 2958 MWt rather than 2900 MWt (2% increase), the short lived iodines have decreased with increased burnup as expected.

Fuel Handling Accident Source Term

The fuel handling accident source term is a fraction of the core source term and represents the gap activity contained in one or more fuel assemblies which has been decayed for 100 hours. This activity is released from the damaged fuel assembly following a fuel handling accident. All of the phenomena affecting the core activity also affect the fuel handling accident source term in addition to FHA specific parameters. These parameters, along with the assumed gap activity, are presented in Table 3.8.2-5 for VANTAGE 5, VANTAGE+, and a generic 2900 MWt core.

The VANTAGE+ activity is substantially greater than the VANTAGE 5 activity. The increase in activity is primarily due to the increased power level and the number of damaged assemblies. The gap fraction, with the exception of the VANTAGE+ I-131 value, is suggested by the NRC in Reference 2. The VANTAGE+ I-131 gap fraction is increased from 0.1 to 0.12 (Reference 3).

Reactor Coolant Activity

Reactor coolant activity for VANTAGE 5, VANTAGE+, and a generic 2900 MWt core are shown in Table 3.8.2-6. In general, the phenomena which affect core activity also affect coolant activity. A parameter which only affects coolant activity is the letdown flow rate. Seventy-five gpm was assumed for the generic plant and 60 gpm was assumed for VANTAGE 5 and VANTAGE+. The decrease in letdown flow results in a decrease in the non-radioactive decay removal terms which results in somewhat higher coolant activities, particularly for the longer half-life isotopes.

TABLE 3.8.2-1

ACTIVITY RELEASED FROM THE CORE FOR A TID-14844 RELEASE (LOCA SOURCE TERM) VANTAGE+ FUEL

Nuclide	Core Activity, Ci
I-131	4.1E7
I-132	6.0E7
I-133	8.4E7
I-134	9.0E7
I-135	7.7E7
Kr-83m	9.5E6
Kr-85m	2.1E7
Kr-85	8.3E5
Kr-87	3.8E7
Kr-88	5.4E7
Kr-89	6.6E7
Xe-131m	5.6E5
Xe-133m	2.4E7
Xe-133	1.7E8
Xe-135m	3.4E7
Xe-135	3.7E7
Xe-138	1.3E8

TID RELEASE: 50% core iodine, 100% core noble gas

TABLE 3.8.2-2

FUEL HANDLING ACCIDENT SOURCE TERM
ACTIVITY RELEASED FROM DAMAGED VANTAGE+ FUEL

Nuclide	Gap Curies
I-131	9.6E4
I-132	6.6E4
I-133	7.8E3
I-135	5.3
Kr-85	4.4E3
Xe-131m	7.6E2
Xe-133m	1.3E4
Xe-133	1.5E5
Xe-135m	8.1E-1
Xe-135	2.7E2

TABLE 3.8.2-3

REACTOR COOLANT ACTIVITY FOR VANTAGE+ FUEL

Nuclide	Reactor Coolant Activity (μCi/gram)
I-131	3.0
I-132	3.1
I-133	4.6
I-134	0.6
I-135	2.4
Kr-85	7.6
Kr-85m	1.8
Kr-87	1.1
Kr-88	3.2
Xe-131m	2.3
Xe-133	2.9E2
Xe-133m	1.9E1
Xe-135	8.6
Xe-135m	0.52
Xe-138	0.64

TABLE 3.8.2-4

COMPARISON OF ACTIVITY RELEASED FROM THE CORE
FOR A TID-14844 RELEASE AND
SALIENT CORE PARAMETERS

	VANTAGE 5	VANTAGE +	Generic
Core Thermal Power, MWt	2775	2958	2900
Average Discharge Burnup, MWD/MTU	41,000	65,370	34,580
Cycle Length, EFPD	428	480	288
Enrichment (wt-% U-235)	3.6	5.0	3.6
Nuclide/Core Activity, Ci			
I-131	3.9E7	4.1E7	4.0E7
I-132	5.6E7	6.0E7	5.8E7
I-133	7.9E7	8.4E7	8.3E7
I-134	8.5E7	9.0E7	9.0E7
I-135	7.3E7	7.7E7	7.7E7
Kr-83m	9.1E6	9.5E6	9.9E6
Kr-85m	2.0E7	2.1E7	2.2E7
Kr-85	6.4E5	8.3E5	5.2E5
Kr-87	3.7E7	3.8E7	4.1E7
Kr-88	5.3E7	5.4E7	5.8E7
Kr-89	6.5E7	6.6E7	7.2E7
Xe-131m	5.4E5	5.6E5	5.6E5
Xe-133m	2.3E7	2.4E7	2.3E7
Xe-133	1.5E8	1.7E8	1.6E8
Xe-135m	3.1E7	3.4E7	3.3E7
Xe-135	3.3E7	3.7E7	3.4E7
Xe-138	1.3E8	1.3E8	1.4E8

TID RELEASE: 50% core iodine, 100% core noble gas

TABLE 3.8.2-5

COMPARISON OF FUEL HANDLING ACCIDENT SOURCE TERMS AND SALIENT PARAMETERS

	VANTAGE 5	VANTAGE +	Generic
Core Thermal Power, Mwt	2775	2958	2900
Radial Peaking Factor	1.65	1.7	1.65
Number of Damaged Assemblies	1.0 (263 rods)	1.2 (314 rods)	1.2 (314 rods)
Decay time, hours	100	100	100
Gap Fraction:			
All Iodine and noble gas except I-131 and Kr-85	0.1	0.1	0.1
I-131	0.1	0.12	0.1
Kr-85	0.3	0.3	0.3
Nuclide/Gap Curies			
I-131	5.8E4	9.6E4	7.5E4
I-132	4.9E4	6.6E4	6.3E4
I-133	6.1E3	7.8E3	7.5E3
I-135	4.0E0	5.3	5.1
Kr-85	2.0E3	4.4E3	2.7E3
Xe-131m	5.5E2	7.6E2	7.0E2
Xe-133m	1.0E4	1.3E4	1.3E4
Xe-133	1.2E5	1.5E5	1.5E5
Xe-135m	6.3E-1	8.1E-1	7.8E-1
Xe-135	2.1E2	2.7E2	2.6E2

TABLE 3.8.2-6

COMPARISON OF REACTOR COOLANT FISSION PRODUCT ACTIVITY

	VANTAGE 5	VANTAGE+	GENERIC
Nuclide		Reactor Coolant Activity, μCi/gram	
I-131	2.8	3.0	2.9
I-132	2.9	3.1	3.0
I-133	4.6	4.6	4.3
I-134	0.67	0.6	0.6
I-135	2.6	2.4	2.3
Kr-85	9.1	7.6	7.7
Kr-85m	2.0	1.8	2.1
Kr-87	1.2	1.1	1.3
Kr-88	3.6	3.2	3.6
Xe-131m	2.2	2.3	2.3
Xe-133	2.6E2	2.9E2	2.8E2
Xe-133m	1.7E1	1.9E1	1.8E1
Xe-135	8.0	8.6	7.7
Xe-135m	0.55	0.52	0.5
Xe-138	0.72	0.64	0.67

3.8.3 Radiological Consequences

The environmental consequences of the FSAR Chapter 15 accidents (Reference 4) have been calculated taking into account the impact of the RSGs, changes in design power capability parameters, and revised source terms presented in Section 3.8.2. All calculations have been performed using the current FSAR methodology except as noted in the sections on each accident. This section presents the results of the limiting dose calculations. In all cases, the dose results are within 10CFR100 limits.

3.8.3.1 Loss of Offsite Power

A postulated loss of offsite power and the subsequent leakage of steam from the secondary system will not result in the release of radioactivity unless there is leakage of reactor coolant to the secondary system within the steam generators. A conservative analysis of the potential offsite doses considering primary to secondary leakage from this accident is performed.

This analysis incorporates source term assumptions based upon one percent failed fuel reactor coolant activity concentrations and a 1 gpm steam generator leakage for sufficient time prior to the accident to establish equilibrium specific activity levels in the secondary side. Source terms from Section 3.8.2 are used as input to the analysis. The method of analysis, the input parameters, and the assumptions used in this calculation are the same as those presented in Section 15.2.9.4 of the VCSNS FSAR with the following exceptions:

1. The mass release of steam from the three steam generators has been revised as follows:

447,900 lbs	(0-2 hrs.)
868,300 lbs	(2-8 hrs.)

2. The feedwater flow to the three steam generators has been revised as follows:

375,500 lbs	(0-2 hrs.)
841,800 lbs	(2-8 hrs.)

3. The steam generator blowdown flowrate is assumed to be a maximum of 10 gpm per RSG.

The offsite radiation doses at the site boundary and the LPZ, resulting from a conservative analysis of a postulated loss of offsite power, are presented in Table 3.8.3-1. The doses for this accident are well within the limits defined in 10CFR100.

3.8.3.2 Waste Gas Decay Tank Rupture

The environmental consequences of a waste gas decay tank rupture have been re-evaluated. While not a function of RSG or changes in the design power capability parameters, the analysis is updated to reflect Technical Specification limits on decay tank radioactivity.

A conservative analysis is performed. The evaluation of the radiation doses resulting from the postulated rupture of a gas decay tank is based on the following assumptions:

1. The quantity of radioactivity contained in a single waste gas decay tank is limited to less than or equal to 160,000 curies noble gases considered as Xe-133 as specified in VCSNS Technical Specification 3.11.2.6. This Technical Specification inventory, which is used to evaluate offsite dose, bounds the conservative noble gas inventory generated by the primary system that was previously considered under the current licensing basis analysis.

2. The decay tank rupture is assumed to occur immediately after isolation of the decay tank from the gaseous waste processing system, releasing the entire contents of the tank to the outside atmosphere at ground level. The assumption of the release of the noble gas inventory from only a single tank is based upon the fact that the valving of the decay tanks in the gaseous waste processing system has been designed so that a release from one decay tank by any means does not result in any additional release of radioactivity stored in any of the other decay tanks.

3. The 0-2 hour accident atmospheric diffusion factor given in FSAR Section 15A (EAB, 4.08E-04 Sec/M^3; LPZ, 1.01E-04 Sec/M^3) is applicable for the conservative offsite dose analysis.

The site boundary gamma and beta doses for the conservative analysis are 0.49 rem and 2.45 rem, respectively.

The low population zone gamma and beta doses for the conservative analysis are 1.21E-1 rem and 6.06E-1 rem respectively.

No significant iodine radioisotope contents within the gas decay tanks are expected; thus, a thyroid inhalation dose is not calculated.

A comparison of the current licensing basis Waste Gas Decay Tank Rupture doses and those calculated herein for the Steam Generator Replacement are given in Table 3.8.3-2. The calculated doses are well within 10CFR100 limits.

3.8.3.3 Break in a CVCS Line

There are no instrument lines connected to the RCS that penetrate containment. However, grab sample lines from the reactor coolant loop 2 and loop 3 hot legs and from the pressurizer steam and liquid spaces and the three-inch CVCS letdown line do penetrate containment. The grab sample lines are equipped with normally-closed isolation valves both inside and outside containment and are designed in accordance with General Design Criteria 55.

The most severe pipe rupture, with regard to the release of radioactivity during normal plant operation, would be the complete severance of the three-inch CVCS letdown line just outside containment upstream of the outer containment isolation valve at rated power. Complete severance of the letdown line would result in the loss of reactor coolant at the rate of 120 gpm. Since this rate of loss is within the capability of the reactor makeup system, engineered safeguard features actuation would not occur.

A postulated break of the CVCS letdown line is considered. Two conservative analyses of the potential offsite doses from this accident are performed. For the first analysis, the concentration of radioactive nuclides in the reactor coolant are based upon plant operation with one percent failed fuel. The second analysis considers the source terms determined in the first analysis and assumes an iodine spike occurring as a result of the reactor shutdown or depressurization of the primary system. The spike is modeled by increasing the equilibrium fission product activity release rate from the fuel by a factor of 500.

Source terms are taken from Section 3.8.2. Except as noted above, the method of analysis, the input parameters, and the assumptions used in this calculation are the same as those presented in Section 15.3.7 of the VCSNS FSAR for this accident.

The offsite radiation doses at the site boundary and the LPZ, resulting from a conservative analysis of a postulated loss of offsite power, are presented in Table 3.8.3-3. Current FSAR results are also presented. The calculated results are well within the limits of 10CFR100.

3.8.3.4 Large Break LOCA

The environmental consequences and control room dose for the postulated LOCA have been performed. The Large Break LOCA is postulated to result in fuel failures and the subsequent release of fission product activity to the containment. The LOCA environmental consequences analysis is performed using the methods detailed in FSAR Section 15.4.1.4. Descriptions of the postulated accident scenarios; source term basis; removal, transport, and release mechanisms; and the resultant offsite and control room doses are presented below.

The assumptions and parameters used to evaluate the offsite and control room doses resulting from the large break LOCA accident analyses are summarized in Table 3.8.3-4. Radioactive releases to the environment emanate from containment and recirculation loop leakage sources. The containment sources are reduced by the action of engineered safeguards equipment whereas the recirculation loop leakages are assumed to be released with no credit for holdup or filtration by the Auxiliary Building HEPA/charcoal filter system. The source term basis for these releases is provided in Section 3.8.2 based upon the parameters given in Table 3.8.3-4.

The thyroid and whole body doses at the site boundary and low population zone are computed in accordance with Regulatory Guide 1.4 (Reference 5) and are summarized in Table 3.8.3-5 along with current FSAR results. The calculated doses are well within 10CFR100 limits.

The post-LOCA control room doses are provided in Table 3.8.3-6 and satisfy the habitability requirements of General Design Criteria 19. Inspection of the results indicate that the calculated offsite doses are also well within 10CFR100 limits.

3.8.3.5 Main Steam Line Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generators. A conservative analysis of the potential offsite doses resulting from a steam line break outside containment is presented assuming a pre-existing 1.0 gpm primary to secondary system steam generator tube leak.

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

1. Two conservative analyses scenarios are evaluated. They are:

   a. A pre-existing iodine spike is assumed to have occurred prior to the steam line break event. Reactor coolant iodine specific activities are assumed to be at the Technical Specification Figure 3.4-1 full power limit of 60 μCi/gm Dose Equivalent Iodine-131. Reactor Coolant noble gas specific activities are conservatively based on 60 times the 1% failed fuel values given in Section 3.8.2, Table 3.8.2-4. The secondary coolant iodine specific activity is conservatively based on secondary coolant specific activity equilibrium being reached with the reactor coolant iodine specific activity at 60 μCi/gm Dose Equivalent Iodine-131, with Technical Specification 3.4.6.2 primary to secondary system tube leakage of 1.0 gpm, and with a minimum steam generator blowdown rate of 30 gpm total for 3 steam generators. The main steam line break event is assumed to result in no failed fuel and consequently no additional release of the fuel gap inventory to the reactor coolant.

   b. A concurrent iodine spike with the steam line break scenario is also postulated. For this scenario, the reactor coolant specific activity is assumed to be at the Technical Specification 3.4.8 normal operation limit of 1.0 μCi/gm Dose Equivalent Iodine-131 for iodines and at the Section 3.8.2, Table 3.8.2-3 (1% failed fuel) specific activity for the noble gases. The secondary system specific activity is assumed to be at the Technical Specification 3.7.1.4 normal operation limit of 0.1 μCi/gm Dose Equivalent I-131. Additionally, concurrent with the MSLB, an iodine spike is assumed to occur which releases iodines from the fuel gap to the reactor coolant at a rate (in Curies/min) of 500 times the normal iodine release rate. The main steam line break event is assumed to result in no failed fuel and consequently no additional release of fuel gap inventory to the reactor coolant is postulated.

2. The 1.0 gpm primary to secondary tube leakage is conservatively assumed to be in the faulted steam generator until isolation and in the intact SG thereafter to maximize offsite dose.

3. The iodine partition factor in the faulted steam generator is assumed to be 1.0 for the steam line break duration. The iodine partition factor in the intact steam generators is assumed to be 0.01 for the steam line break duration. The iodine partition factor is defined as follows:

$$\frac{\text{amount of iodine / unit mass steam}}{\text{amount of iodine / unit mass liquid}}$$

4. No noble gas is dissolved or contained in the secondary system water, i.e., all noble gas leaked to the secondary system is continuously released from the steam generators and secondary system.

5. Offsite power is lost and the main condenser is not available for steam dump.

6. Eight hours after the accident, the Residual Heat Removal System starts operation to cool down the plant. After 8 hours, no further steam or activity release occurs due to the steam line break event.

7. Dilution effects of incoming feedwater flow to the intact steam generators are not considered.

8. The 0-2 hour and 2-8 hour accident atmospheric diffusion factors are used. For the EAB, X/Q = 4.08E-04 Sec/M^3 for the 0-2 hours period. For the LPZ, X/Q = 1.01E-04 Sec/M^3 and 2.37E-05 Sec/M^3 for the 0-2 and 2-8 hours periods, respectively.

Steam releases to the atmosphere, due to the postulated MSLB, are:

Faulted SG:	0-30	Minutes:	406,000 lbm
	0.5-8	Hours:	0 lbm
Intact SGs:	0-2	Hours:	343,700 lbm
	2-8	Hours:	733,900 lbm

The main steam line break conservative case results (with iodine spikes) for gamma, beta, and thyroid inhalation doses are given in Table 3.8.3-7. A comparison to current FSAR results is also provided. For both cases, the radiological consequences are well within 10CFR100 limits.

3.8.3.6 Steam Generator Tube Rupture

A conservative analysis of the potential offsite doses resulting from a postulated SGTR is presented assuming a 1.0 gpm primary to secondary system steam generator tube leak existed prior to and during the postulated tube rupture event.

The following assumptions and parameters are used to calculate the conservative activity releases and offsite doses for a steam generator tube rupture:

1. Two conservative analyses scenarios are evaluated. They are:

    a. A pre-existing iodine spike is assumed to have occurred prior to the steam generator tube rupture event. Reactor coolant iodine specific activities are assumed to be at the Technical Specification Figure 3.4-1 full power limit of 60 μCi/gm Dose Equivalent Iodine-131. Reactor Coolant noble gas specific activities are conservatively based on 60 times the 1% failed fuel values given in Section 3.8.2, Table 3.8.2-4. The secondary coolant iodine specific activity is conservatively based on secondary coolant specific activity equilibrium being reached with the reactor coolant iodine specific activity at 60 μCi/gm Dose Equivalent Iodine-131, with Technical Specification 3.4.6.2 primary to secondary system tube leakage of 1.0 gpm, and with a minimum steam generator blowdown rate of 30 gpm total for 3 steam generators. The steam generator tube rupture event is assumed to result in no failed fuel and consequently no additional release of the fuel gap inventory to the reactor coolant.

    b. A concurrent iodine spike with the steam generator tube rupture scenario is also postulated. For this scenario, the reactor coolant specific activity is assumed to be at the Technical Specification 3.4.8 normal operation limit of 1.0 μCi/gm Dose Equivalent Iodine-131 for iodines and at the Section 3.8.2, Table 3.8.2-3 (1% failed fuel) specific activity for the noble gases. The secondary system specific activity is assumed to be at the Technical Specification 3.7.1.4 normal operation limit of 0.1 μCi/gm Dose Equivalent I-131. Additionally, concurrent with the steam generator tube rupture event, an iodine spike is assumed to occur which releases iodines from the fuel gap to the reactor coolant at a rate (in Curies/min) of 500 times the normal iodine release rate. The steam generator tube rupture event is assumed to result in no failed fuel and consequently no additional release of fuel gap inventory to the reactor coolant is postulated.

2. The 1.0 gpm primary to secondary tube leakage is all assumed to be in the intact steam generators for the accident duration.

3. The iodine partition factor in the faulted steam generator is assumed to be 0.10 for the steam generator tube rupture accident till the faulted SG is isolated. The iodine partition factor in the intact steam generators is assumed to be 0.01 for all iodines. The iodine partition factor is defined as follows:

$$\frac{\text{amount of iodine / unit mass steam}}{\text{amount of iodine / unit mass liquid}}$$

4. No noble gas is dissolved or contained in the secondary system water, i.e., all noble gas leaked to the secondary system is continuously released from the steam generators and secondary system.

5. Offsite power is lost and the main condenser is not available for steam dump.

6. Eight hours after the accident, the Residual Heat Removal System starts operation to cool down the plant. After eight hours, no further steam or activity release occurs due to the steam line break event.

7. Dilution effects of incoming feedwater flow to the intact steam generators are not considered.

8. The 0-2 hour and 2-8 hour accident atmospheric diffusion factors are used. For the EAB, X/Q = 4.08E-04 Sec/M^3 for the 0-2 hours period. For the LPZ, X/Q = 1.01E-04 Sec/M^3 and 2.37E-05 Sec/M^3 for the 0-2 and 2-8 hours periods, respectively.

Primary system mass released via the tube rupture is 92,900 lbm.

Steam releases to the atmosphere, due to the postulated SGTR, are (see Table 3.5-2):

Faulted SG:	0-30	Minutes:	56,800 lbm
	0-8	Hours:	0 lbm
Intact SGs:	0-2	Hours:	381,400 lbm
	2-8	Hours:	924,900 lbm

The steam generator tube rupture conservative case results (with iodine spike) for gamma, beta, and thyroid inhalation doses are given in Table 3.8.3-8. Current FSAR results are also presented. The calculated doses are well within 10CFR100 limits.

3.8.3.7 Locked Rotor

A postulated reactor coolant pump locked rotor event and the subsequent leakage of steam from the secondary system, due to the leakage of reactor coolant to the secondary system within the steam generators is considered. This analysis assumes steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. A conservative analysis of the potential offsite doses from this accident is performed with the steam generator leakage as a variable parameter.

Source terms from Section 3.8.2, assuming 1% defective fuel prior to the accident and 15% additional fuel failure as a result of the locked rotor, are used in the analysis. The input parameters, and the assumptions used in this calculation are the same as those presented in Section 15.4.4.4 of the VCSNS FSAR for this accident with the following exceptions:

1. The mass release of steam from the three steam generators has been revised as follows:

<u>Steam Releases from three RSGs</u>	447,900 lbm (0-2 hrs)
	868,300 lbm (2-8 hrs)

2. The feedwater flow to the three steam generators has been revised as follows:

<u>Feedwater Delivery to three RSGs</u>	375,500 lbm (0-2 hrs)
	841,800 lbm (2-8 hrs)

3. The steam generator blowdown flow rate is assumed to be 10 gpm per RSG.

The offsite radiation doses at the site boundary and the LPZ, resulting from this conservative analysis of a locked rotor, are presented in Table 3.8.3-9. Even with the overly conservative assumption of 15% fuel failure, the calculated results are well within the limits of 10CFR100.

3.8.3.8 Fuel Handling Accident

An FHA during refueling could result in the release of a fraction of the fission product inventory in the plant to the environment. Two accident scenarios are considered: a refueling accident occurring inside containment, and a refueling accident occurring outside containment. A conservative analysis is performed for both cases.

The postulated FHA inside containment is the dropping of a spent fuel assembly onto the core during refueling which results in damage to the fuel assemblies. Following the postulated accident inside the containment, the activity released to the reactor building atmosphere is assumed to be instantaneously released to the environment through the Reactor Building Purge System. In the analysis, no credit is taken for a reduction in the amount of activity released due to filtration or radioactive decay due to holdup in the containment. Source terms are taken from Section 3.8.2. The method of analysis, the input parameters, and the assumptions used in this calculation are the same as those presented in Section 15.4.5.1 of the FSAR for this accident with the following exceptions:

1. The radial peaking factor has been increased from 1.65 to 1.70.

2. The clad gap activity for I-131 has been increased from 10% to 12%.

The postulated FHA outside containment is the dropping of a spent fuel assembly onto the Spent Fuel Pool which results in damage to the fuel assemblies and the release of the volatile gaseous fission products. Following the postulated accident outside the containment, the activity released to the Spent Fuel Pool is subsequently released to the Fuel Handling Building and then released to the environment through the Fuel Handling Building Charcoal Exhaust System. In the analysis, no credit is taken for the mixing of the activity released with the Fuel Building atmosphere nor for radioactive decay due to holdup in the building or transit time after release to the environs. A charcoal filter efficiency of 95% is assumed for all forms of iodine released from the building. Appropriate source terms are taken from Section 3.8.2. The method of analysis, the input parameters, and the assumptions used in this calculation are the same as those presented in Section 15.4.5.2 of the VCSNS FSAR for this accident, with the exceptions described above.

The offsite radiation doses at the site boundary and the LPZ, resulting from a conservative analysis of a postulated FHA inside and outside of containment, are presented in Table 3.8.3-10 along with the current FSAR results. The calculated doses are within 10CFR100 limits.

3.8.3.9 RCCA Ejection

An analysis is performed for a postulated rod ejection accident. It is assumed that the plant is operating at equilibrium levels of radioactivity in the primary and secondary systems prior to the postulated accident as a result of coincident fuel defects (1%) and steam generator tube leakage (1 gpm). Following a postulated rod ejection accident, two potential activity release pathways contribute to the total radiological consequences of the accident. The first path is via containment leakage of activity released to the containment from the reactor coolant and is the only significant contributor. The second pathway is via the contaminated steam from the secondary system which is released through the relief valves since it is assumed that offsite power is lost. The potential offsite doses from this accident are calculated, based on Regulatory Guide 1.77 (Reference 6).

Source terms from Section 3.8.2, assuming 1% defective fuel prior to the accident and 10% additional fuel failure as a results of the RCCA Ejection, are used in the analysis. The method of analysis, the input parameters, and the assumptions used in this calculation are the same as those presented in Section 15.4.6.4 of the FSAR for this accident.

The offsite radiation doses at the site boundary and the LPZ, resulting from an analysis of a RCCA Ejection are presented in Table 3.8.3-11. Current FSAR results are also presented. The calculated results are within 10CFR100 limits.

TABLE 3.8.3-1

LOSS OF OFFSITE POWER
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSG (Rem)
Conservative Case:			
EAB:	Gamma:	6.65E-4	6.9E-4
	Beta:	1.30E-3	1.3E-3
	Thyroid:	2.87E-2	4.9E-2
LPZ:	Gamma:	1.53E-4	1.6E-4
	Beta:	3.00E-4	3.2E-4
	Thyroid:	3.95E-3	8.3E-3

TABLE 3.8.3-2

WASTE GAS DECAY TANK RUPTURE
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSGs (Rem)
Conservative Case:			
EAB:	Gamma:	1.16E-1	4.90E-1
	Beta:	3.22E-1	2.45E+0
	Thyroid:	4.02E-2	(1)
LPZ:	Gamma:	2.87E-2	1.21E-1
	Beta:	8.04E-2	6.06E-1
	Thyroid:	9.95E-3	(1)

(1) Not calculated.

TABLE 3.8.3-3

BREAK IN A CVCS INSTRUMENT LINE
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)		Proposed Licensing Basis with RSGs (Rem)
Conservative Case:				
EAB:	Gamma:	2.85E-2	No Iodine Spike:	1.8E-2
			Concurrent Iodine Spike:	2.5E-2
	Beta:	3.20E-2	No Iodine Spike:	3.4E-2
			Concurrent Iodine Spike:	3.7E-2
	Thyroid:	7.60E-2	No Iodine Spike:	8.0E-1
			Concurrent Iodine Spike:	3.0E+0
LPZ:	Gamma:	1.66E-3	No Iodine Spike:	1.1E-3
			Concurrent Iodine Spike:	1.5E-3
	Beta:	1.86E-3	No Iodine Spike:	2.0E-3
			Concurrent Iodine Spike:	2.2E-3
	Thyroid:	4.41E-2	No Iodine Spike:	4.6E-2
			Concurrent Iodine Spike:	1.8E-1

Note: The current FSAR analysis did not include the effects of a concurrent iodine spike.

TABLE 3.8.3-4

PARAMETERS USED TO EVALUATE A LARGE BREAK LOCA

DESCRIPTION/PARAMETER	ASSUMPTION/VALUE
SOURCE TERMS	
CORE THERMAL POWER	2958 MWt
ACTIVITY RELEASED TO RB:	
CORE INVENTORY:	
Iodines	50%
Noble Gases	100%
ACTIVITY RELEASED TO RB SUMP:	
Iodines	50%
Noble Gases	0
Flashing Fraction	0.1
IODINE PLATEOUT INSIDE RB	50%
IODINE SPECIES:	
Elemental	91%
Particulate	5%
Organic	4%
REACTOR BUILDING:	
FREE VOLUME (ft^3)	1.84E+6
LEAKAGE RATE (per day)	0.2% (0-24 hr) 0.1% (1-30 days)
SUMP LIQUID VOLUME (ft^3)	5.83E+4
OUTER WALL CONCRETE THICKNESS (ft)	4
RB COOLING SYSTEM:	
RECIRCULATION FLOW RATE (cfm)	5.42E+4
RECIRCULATION FILTER EFFICIENCY:	
Elemental Iodine	0%
Particulate Iodine	90%
Organic Iodine	0%

TABLE 3.8.3-4 (Continued)

PARAMETERS USED TO EVALUATE A LARGE BREAK LOCA

DESCRIPTION/PARAMETER	ASSUMPTION/VALUE
RB SPRAY SYSTEM:	
OPERATING SPRAY PUMPS	1 of 2
POST LOCA ACTUATION TIME (sec)	52
IODINE SPRAY REMOVAL CONSTANTS:	
Elemental (hr^{-1})	20.33
Particulate (hr^{-1})	5.680 (0 - 98% removal) 0.568 (98 - 100% removal)
MAXIMUM DF ON ELEMENTAL IODINE	100
RECIRCULATION LOOP:	
OPERATIONAL LEAKAGE (cc/hr)	5860
PASSIVE COMPONENT FAILURE (gpm) (for 30 min/24 hrs post LOCA)	50
MINIMUM TIME TO RECIRC MODE (sec)	2335
CONTROL ROOM:	
FREE VOLUME (ft^3)	226040
OUTER WALL/ROOF CONCRETE THICKNESS (ft)	2
FILTERED RECIRCULATION FLOW (cfm)	19143
RECIRCULATION FILTER EFFICIENCY (for all forms of iodine)	95%
FILTERED AIR INFILTRATION RATE (cfm)	
Technical Specification Limit	1000
Maximum based on GDC 19	2665
UNFILTERED AIR INFILTRATION RATE (cfm)	10

TABLE 3.8.3-4 (Continued)

PARAMETERS USED TO EVALUATE A LARGE BREAK LOCA

DESCRIPTION/PARAMETER	ASSUMPTION/VALUE
ENVIRONMENTAL/HUMAN FACTORS:	
DISTANCES:	
Exclusion Area Boundary (mile)	1
Low Population Zone (miles)	3
Reactor Building to Control Building (ft)	200
ATMOSPHERIC DISPERSION FACTORS:	
OFFSITE X/Q VALUES (sec/m^3)	
EXCLUSION BOUNDARY	
0 - 1 hour	4.08E-4
0 - 8 hours	8.43E-5
8 - 24 hours	1.34E-5
1 - 4 days	6.12E-6
4 - 30 days	3.52E-6
LOW POPULATION ZONE	
0 - 8 hours	2.37E-5
8 - 24 hours	2.44E-6
1 - 4 days	1.11E-6
4 - 30 days	6.28E-7
CONTROL ROOM X/Q VALUES (sec/m^3)	
0 - 8 hours	9.35E-4
8 - 24 hours	6.63E-4
1 - 4 days	3.95E-4
4 - 30 days	2.45E-4
BREATHING RATES:	
OFFSITE (m^3/sec)	
0 - 8 hours	3.47E-4
8 - 24 hours	1.75E-4
1 - 30 days	2.32E-4
CONTROL ROOM (m^3/sec)	
0 - 30 days	3.47E-4
CONTROL ROOM OCCUPANCY FACTORS	
0 - 8 hours	1.0
8 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4
METHOD OF DOSE CALCULATION	FSAR Appendix 15A

TABLE 3.8.3-5

LARGE BREAK LOCA
OFFSITE DOSE COMPARISON - REGULATORY GUIDE 1.4 ANALYSIS

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSG (Rem)
Conservative Case:			
EAB:	Gamma:	2.78E+0	3.07E+0
	Beta:	2.16E+0	2.13E+0
	Thyroid:	1.74E+2	1.10E+2
LPZ:	Gamma:	3.57E-1	3.66E-1
	Beta:	3.07E-1	3.15E-1
	Thyroid:	2.89E+1	3.03E+1

TABLE 3.8.3-6

LARGE BREAK LOCA
CONTROL ROOM DOSE COMPARISON

Control Room Dose:	Current Licensing Basis (Rem)	Proposed Licensing Basis with RSG (Rem)	
Conservative Case:			
Note:	(1)	(2)	(3)
Gamma:	1.70E+0	1.60E+0	1.32E+0
Beta:	6.60E+0	6.90E+0	5.50E+0
Thyroid:	3.00E+1	3.00E+1	1.40E+1

Notes:

1. Based upon a maximized control room filtered inleakage rate of 2413 cfm.
2. Based upon a maximized control room filtered inleakage rate of 2665 cfm.
3. Based upon Technical Specification limit, 1000 cfm, of filtered inleakage/makeup air into the control room.

TABLE 3.8.3-7

MAIN STEAM LINE BREAK
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSGs (Rem)
Conservative Case:			
EAB:	Gamma:	1.23E-2	Pre-existing Iodine Spike:4.88E-2
			Concurrent Iodine Spike:3.90E-3
	Beta:	8.04E-3	Pre-existing Iodine Spike:8.20E-2
			Concurrent Iodine Spike:2.71E-3
	Thyroid:	1.27E+1	Pre-existing Iodine Spike:1.49E+1
			Concurrent Iodine Spike:4.05E+0
LPZ:	Gamma:	2.70E-3	Pre-existing Iodine Spike:1.89E-2
			Concurrent Iodine Spike:1.12E-3
	Beta:	1.78E-3	Pre-existing Iodine Spike:3.38E-2
			Concurrent Iodine Spike:9.11E-4
	Thyroid:	2.76E+0	Pre-existing Iodine Spike:3.75E+0
			Concurrent Iodine Spike:1.02E+0

Note: The current FSAR conservative analysis did not include the effects of pre-existing and concurrent iodine spikes. However, the current FSAR analysis assumes 5% fuel failure resulting from the MSLB, even though fuel failure is not predicted. No fuel failure is calculated for the MSLB accident when incorporating the effects of the RSGs and revised design power capability parameters.

TABLE 3.8.3-8

STEAM GENERATOR TUBE RUPTURE
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSGs (Rem)
Conservative Case:			
EAB:	Gamma:	1.88E-1	Pre-existing Iodine Spike:5.27E+0
			Concurrent Iodine Spike:1.01E-1
	Beta:	2.17E-1	Pre-existing Iodine Spike:1.04E+1
			Concurrent Iodine Spike:1.77E-1
	Thyroid:	4.13E-1	Pre-existing Iodine Spike:5.35E+1
			Concurrent Iodine Spike:5.04E+0
LPZ:	Gamma:	4.71E-2	Pre-existing Iodine Spike:1.31E+0
			Concurrent Iodine Spike:2.52E-2
	Beta:	5.46E-2	Pre-existing Iodine Spike:2.58E+0
			Concurrent Iodine Spike:4.42E-2
	Thyroid:	3.19E-1	Pre-existing Iodine Spike:1.33E+1
			Concurrent Iodine Spike:1.26E+0

Note: The current FSAR conservative analysis did not include the effects of pre-existing and concurrent iodine spikes.

TABLE 3.8.3-9

LOCKED ROTOR
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSGs (Rem)
Conservative Case:			
EAB:	Gamma:	5.63E-1	1.2E+0
	Beta:	8.27E-1	8.1E-1
	Thyroid:	8.01E+0	7.8E+0
LPZ:	Gamma:	1.31E-1	2.8E-1
	Beta:	1.92E-1	1.9E-1
	Thyroid:	1.85E+0	1.8E+0

Note: These results are based on a primary to secondary leak rate of 1.0 gpm.

TABLE 3.8.3-10

FUEL HANDLING ACCIDENT
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSGs (Rem)
Inside Containment			
Conservative Case:			
EAB:	Gamma:	1.40E+0	7.08E-1
	Beta:	1.65E+0	1.78E+0
	Thyroid:	1.53E+2	2.11E+2
LPZ:	Gamma:	(1)	4.11E-2
	Beta:	(1)	1.03E-1
	Thyroid:	(1)	1.23E+1
Outside Containment			
Conservative Case:			
EAB:	Gamma:	1.40E+0	5.20E-1
	Beta:	1.65E+0	1.72E+0
	Thyroid:	7.66E+0	1.06E+1
LPZ:	Gamma:	(1)	3.02E-2
	Beta:	(1)	9.99E-2
	Thyroid:	(1)	6.16E-1

(1) Not calculated.

TABLE 3.8.3-11

RCCA EJECTION
OFFSITE DOSE COMPARISON

Offsite Dose:		Current Licensing Basis (Rem)	Proposed Licensing Basis with RSGs (Rem)
Ultra-Conservative Case:			
EAB:	Gamma:	1.56E-1	1.6E-1
	Beta:	7.20E-2	7.6E-2
	Thyroid:	5.28E+1	5.6E+1
LPZ:	Gamma:	2.36E-2	2.5E-2
	Beta:	1.15E-2	1.2E-2
	Thyroid:	1.46E+1	1.5E+1

3.8.4 References

1. DiNunno, J.J., Anderson, F.D., Baker, R.E., and Waterfield, R.L., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, 3/23/62.

2. USNRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

3. Baker, D.A., et al.,"Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR-5009, February 1988.

4. Virgil C. Summer Nuclear Station Final Safety Analysis Report

5. USNRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

6. USNRC Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

3.9.8 Application of Leak-Before-Break Methodology

In January 1993, the use of the Leak-Before-Break (LBB) methodology as technical justification for eliminating large primary loop pipe rupture as the structural design basis for VCSNS was approved by the NRC (WCAP-13206, Reference 1). Up until that point, the structural design basis for the VCSNS RCS required that the dynamic effects of pipe breaks be evaluated and that protective measures for those breaks be incorporated into the design. The LBB methodology effort demonstrated that the primary loops for VCSNS are highly resistant to stress corrosion cracking and high and low cycle fatigue, and that water hammer is mitigated by system design and operating procedures. The effort was performed initially assuming the current VCSNS configuration (e.g., Model D3 steam generators, current power level and licensed operating conditions).

The LBB evaluation was re-examined as described in WCAP-13605 to incorporate the effects of hardware changes and potential stretch power applications at VCSNS. The hardware changes include removal of SG support snubbers, removal of crossover leg whip restraints and the replacement of the steam generators.

The results of the calculations performed to reconcile the elimination of the RCS primary loop breaks for the VCSNS under the new loop configuration and potential stretch power application demonstrate that the conclusions reached in Reference 1 remain unchanged. The conclusions are listed below and further discussed in WCAP-13605 (see Appendix 5).

- Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation.

- Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations.

- The effects of low and high cycle fatigue on the integrity of the primary piping are negligible.

- Adequate margin exists between the leak rate of small stable flaws and the capability of the VCSNS RCS pressure boundary Leakage Detection System.

- Ample margin exists between the small stable flaw sizes discussed above and larger stable flaws.

- Ample margin exists in the material properties used to demonstrate end-of-service life (relative to aging) stability of the critical flaws.

Based on the above, it is concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis for the VCSNS.

References:

1. WCAP 13206, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Virgil C. Summer Nuclear Power Plant", dated April, 1992, by J. C. Schmertz et al., Westinghouse. (WCAP 13207 is Proprietary Class 3 Version of this report.)

2. WCAP 13605, "Primary Loop Leak-Before-Break Reconciliation to Account for the Effects of Steam Generator Replacement/Uprating", dated March 1993, by Y. C. Lee et al., Westinghouse. (Appendix 5)

APPENDIX 5