Enclosure 6 to SERIAL: HNP-00-142

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

NSSS LICENSING REPORT

BOOK 2 of 2

6.3 Steam Generator Tube Rupture Transient

In support of the Harris Nuclear Plant (HNP) SGR/Uprating program, an evaluation for a design basis steam generator tube rupture (SGTR) event has been performed to demonstrate that the potential consequences are acceptable. The evaluation discussed herein considers operation with a full power average temperature (T_{avg}) of 588.8°F and assumes that up to 10 percent of the steam generator tubes are plugged. The analysis supports a main feedwater temperature window of 375°F to 440°F. Operation with the Model Delta 75 replacement steam generators at the current NSSS power level of 2787.4 MWt was also considered.

The major hazard associated with a SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiation doses resulting from a SGTR are within the allowable guidelines. One of the major concerns for a SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiation doses. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The analysis confirmed that steam generator overfill does not occur. A thermal and hydraulic analysis was also performed to determine the input for use in calculating the offsite radiation doses, assuming the limiting single failure relative to offsite doses without steam generator overfill. The limiting single failure assumptions for these analyses were evaluated.

Plant response to the event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the ruptured steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a SGTR based on the HNP Emergency Operating Procedures, which are based on the Westinghouse Owners Group Emergency Response Guidelines.

The LOFTTR2 analyses were performed for the time period from the SGTR until the primary and secondary pressures are equalized (break flow termination). In the margin to overfill analysis presented in Section 6.3.1, the water volume in the secondary side of the ruptured steam generator was calculated as a function of time to demonstrate that overfill does not occur. The thermal and hydraulic analysis to develop input to the radiological consequences analysis is presented in Section 6.3.2. In this analysis the primary to secondary break flow and the steam releases to the atmosphere from both the ruptured and intact steam generators were calculated for use in determining the activity released to the atmosphere. The mass releases were calculated with the LOFTTR2 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases from and feedwater flows to the intact and ruptured steam generators were determined from a mass and energy balance using the calculated reactor coolant system (RCS) and steam generator conditions at the time of leakage termination. The mass release information is used to calculate the radiation doses at the site boundary and low population zone and to the operators in the control room. The radiological consequences analysis is presented in Section 6.3.3.

6.3.1 Analysis of Margin to Steam Generator Overfill

6.3.1.1 Introduction

The SGTR analyses were performed for HNP using the analysis methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in Safety Evaluation Reports (SERs) dated December 17, 1985 and March 30, 1987. The methodology was developed for use with the LOFTTR2 program, an updated version of the LOFTTR1 program. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in References 1 and 2. This is the same methodology employed in the most recent analyses performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 3 and 4).

An analysis was performed to determine the margin to steam generator overfill for a design basis SGTR event for HNP. The analysis was performed using the LOFTTR2 program and the methodology developed in Reference 1, and using the plant specific parameters for HNP. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

6.3.1.2 Description of Analyses and Evaluations

The margin to overfill analysis assumes that the plant is operating with the feedwater temperature at the low end of the temperature window, since this results in a higher mass of water in the steam generator at the start of the event, which limits the amount of break flow and auxiliary feedwater (AFW) that can accumulate in the ruptured steam generator without forcing water into the steamlines. Maximum (10-percent) tube plugging is assumed in the thermal-hydraulic analysis to determine the margin to overfill since this reduces the heat transfer to the ruptured steam generator, minimizing the amount of mass released from the steam generator due to steaming, which in turn reduces the margin to overfill. The reduced heat transfer also prolongs the cooldown period, leading to delayed break flow termination. Although maximum tube plugging results in a lower initial water mass in the ruptured steam generator, which increases the available margin to overfill at the start of the event, this is not more limiting than the lower heat transfer effects described above. This has been confirmed via sensitivity runs specifically made for the HNP SGR/Uprating program.

Design Basis Accident

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary to secondary leakage than a break on the hot side of the steam generator as determined by Reference 1. It was also assumed that a loss of offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The potentially limiting single failures with respect to margin to steam generator overfill for a SGTR from Reference 1 are outlined below.

1. Auxiliary Feedwater (AFW) Flow Control Valve Failure

   The AFW control valves are normally open and are used to control inventory in the intact steam generators and terminate feedwater flow to the ruptured steam generator. A failure of the ruptured steam generator control valve would require the operator to perform additional action to stop the associated AFW pump in order to terminate AFW flow to the ruptured steam generator. It is assumed that 120 seconds (2 minutes) of operator action time will be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump. This 2 minutes is added to the time for AFW isolation assumed without this failure, which is discussed later in this section and identified in Table 6.3.1-1. The additional AFW provided to the ruptured steam generator during that 2 minutes decreases the margin to overfill. The continued AFW flow also results in a reduction in steaming from the ruptured steam generator, which reduces the margin to overfill.

2. Intact Steam Generator Power Operated Relief Valve (PORV) Failure

   Since offsite power is assumed to be lost at reactor trip for the SGTR analysis, the PORVs are relied upon to cool the reactor coolant system. Failure of a PORV on an intact steam generator to open on demand reduces the steam releases capability to that provided by a single PORV. This increases the time required for the cooldown, resulting in increased break flow and a reduction in the margin to steam generator overfill. As indicated in Reference 1 this is typically the limiting single failure for a three loop plant.

   The analyses performed for the HNP replacement steam generator and uprate programs specifically evaluated these two potentially limiting single failures and determined that the limiting failure for the margin to overfill analysis is the failure of the PORV on one of the two intact steam generators to open for the cooldown. The analysis presented in this report models this failure.

Conservative Assumptions

Plant responses until break flow termination were calculated using the LOFTTR2 computer code. The conservative conditions and assumptions which were used in Reference 1 were also used in the LOFTTR2 analysis to determine margin to steam generator overfill for HNP with the exception of the following differences.

1. Reactor Trip and Turbine Runback

   A turbine runback can either be initiated automatically or the operator can manually reduce the turbine load to attempt to prevent a reactor trip on overtemperature-ΔT. Although turbine runback is simulated in this analysis, credit is not taken for delaying reactor trip. Until reactor trip and the assumed loss of offsite power, the main feedwater control system is assumed to maintain a constant steam generator water level. Therefore, until reactor trip, the break flow does not reduce the margin to overfill. An earlier reactor trip will result in an earlier increase in the ruptured steam generator water volume and earlier initiation of AFW. These effects will result in an increased secondary mass in the ruptured steam generator at the time of isolation since the isolation is assumed to occur at a fixed time after the SGTR occurs rather than at a fixed time after reactor trip. For this analysis the time of reactor trip on overtemperature-ΔT was determined by modeling the HNP protection system to occur at approximately 113 seconds. The effect of turbine runback was simulated until reactor trip at the rate of 10 percent per minute. The effect of turbine runback was conservatively simulated by increasing the secondary mass by the differential between 81 percent and 100 percent power and performing the analysis at 100 percent power.

2. Steam Generator Secondary Mass

   A higher initial secondary water mass in the ruptured steam generator was determined by Reference 1 to be conservative for overfill. As noted above, turbine runback was assumed to be initiated and was simulated by artificially increasing the initial steam generator water mass. The initial steam generator total fluid mass was assumed to be 10 percent above the nominal full power fluid mass, plus the differential mass between 100 percent power and 81 percent power to simulate the effect of turbine runback.

3. AFW System Operation

   For this analysis the maximum AFW flow rate of 500 gpm to the ruptured steam generator was assumed to be initiated immediately after reactor trip with no startup delays credited.

Operator Action Times

In the event of a SGTR, the operator is required to take actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the HNP EOPs PATH-2, and major actions were explicitly modeled in this analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary side activity, as indicated by the condenser vacuum pump effluent radiation monitor, steam generator blowdown line radiation monitor, or main steamline radiation monitor, typically will provide the first indication of a SGTR event. The ruptured steam generator can be identified by a mismatch between steam and feedwater flow, high activity in a steam generator water sample, or a high radiation indication on the corresponding main steamline radiation monitor. For a SGTR that results in a reactor trip at high power as assumed in this analysis, the steam generator water level as indicated on the narrow range will decrease significantly for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of a SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator by (1) minimizing the accumulation of feedwater flow and (2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage. In the HNP EOP for steam generator tube rupture, the operator is directed to maintain the level in the ruptured steam generator between 10 percent and 50 percent on the narrow range instrument. To model the isolation time using the methodology in Reference 1, it was assumed that AFW flow to the ruptured steam generator would be isolated when level in the steam generator reached 30 percent narrow range level or at 10 minutes, whichever is longer. Complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30 percent on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer.

3. Cooldown the Reactor Coolant System (RCS) using the intact steam generator.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the power-operated relief valves (PORVs) on the intact steam generators. Since offsite power is assumed to be lost at reactor trip for this analysis, the cooldown was performed by dumping steam via the PORVs on the intact steam generators. Due to the single failure assumed in this analysis only one intact steam generator is used for the cooldown.

4. Depressurize the RCS to restore reactor coolant inventory.

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

   The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. Since offsite power is assumed to be lost at the time of reactor trip, the RCPs are not running and thus normal pressurizer spray is not available. Therefore, the depressurization is modeled using a pressurizer power operated relief valve (PORV).

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator.

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

The operator action times to identify and isolate the ruptured steam generator, to initiate RCS cooldown, to initiate RCS depressurization, and to perform safety injection termination were developed for the design basis analysis in Reference 1. CP&L has determined the corresponding operator action times to perform these operations for HNP. The operator actions and the corresponding operator action times used for the HNP analysis are listed in Table 6.3.1-1.

These operator action times are different from those modeled in the Reference 3 and 4 analyses. For the replacement steam generator analysis, the operator actions for isolation of AFW flow to the ruptured steam generator and isolation of steam flow from the ruptured steam generator have been separated, while in the Reference 3 and 4 analyses, these actions were performed simultaneously. Both isolation times are earlier than assumed in References 3 and 4. The times from steamline isolation until the cooldown is initiated and from the end of cooldown until the depressurization is initiated have been changed from those assumed in References 3 and 4. The Reference 3 and 4 times are provided in Table 6.3.1-1.

6.3.1.3 Description of Analysis Cases

The LOFTTR2 analysis results for the HNP margin to overfill analysis with Model Delta 75 replacement steam generators and operation at the uprated NSSS power of 2912.4 MWt are described below. The sequence of events for this transient is presented in Table 6.3.1-2.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 6.3.1-1. The RCS pressure also decreases as shown in Figure 6.3.1-2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on an overtemperature-ΔT trip signal at approximately 113 seconds.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 6.3.1-3. As a result of the assumed loss of offsite power, main feedwater flow was assumed to be terminated and AFW flow was assumed to be automatically initiated following reactor trip. The air supply to the MSIVs is assumed to fail on the loss of offsite power and the MSIVs will go to the closed position as the instrument air supply degrades.

The RCS pressure and pressurizer level continue to decrease after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal at approximately 183 seconds. The SI flow increases the reactor coolant inventory and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figure 6.3.1-4); however, the temperature differential subsequently increases as the reactor coolant pumps coast down and natural circulation flow develops. The cold-leg temperature trends toward the steam generator temperature as the fluid residence time in the tube region increases. The RCS temperatures continue to slowly decrease due to the continued addition of the auxiliary feedwater to the steam generators until operator actions are initiated to cool down the RCS.

Major Operator Actions

1. Identify and Isolate the ruptured steam generator.

   Recovery actions begin by throttling the auxiliary feedwater flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously, auxiliary feedwater flow to the ruptured steam generator is assumed to be identified and isolated when the narrow range level reaches 30 percent on the ruptured steam generator or at 10 minutes after initiation of the SGTR, whichever is longer. For the HNP analysis the time to reach 30 percent is less than 10 minutes, and thus the ruptured steam generator is assumed to be isolated at 10 minutes. Also, as indicated previously, complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30 percent on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer. For the HNP analysis the time to reach 30 percent is less than 12 minutes, and thus the ruptured steam generator is assumed to be isolated at 12 minutes.

2. Cooldown the RCS to establish subcooling margin.

   After isolation of the ruptured steam generator, a 5-minute operator action time is imposed prior to initiating the cooldown. After this time, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the PORV on both intact steam generators. However, as noted previously, the limiting single failure was assumed to be the failure of one of the intact steam generator PORVs to lift on demand. It was therefore assumed only one of the intact steam generator's PORV is opened at 17 minutes for the RCS cooldown. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 20°F for subcooling uncertainty. When these conditions are satisfied at 1960 seconds, it is assumed that the operator closes the intact steam generator's PORV to terminate the cooldown. This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure 6.3.1-3, and the effect of the cooldown on the RCS temperature is also shown in Figure 6.3.1-4. The pressurizer level and RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure 6.3.1-1.

3. Depressurize the RCS to restore inventory.

   The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. A 4-minute operator action time is included prior to the RCS depressurization. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The RCS depressurization is initiated at 2200 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 10 percent for pressurizer level uncertainty, or pressurizer level is greater than 75 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated at 2332 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10 percent. The RCS depressurization (Figure 6.3.1-2) reduces the break flow as shown in Figure 6.3.1-5 and increases SI flow to refill the pressurizer, as shown in Figure 6.3.1-1.

4. Terminate SI to stop primary to secondary leakage.

   The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 20°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 10-percent allowance for uncertainty.

   After depressurization is completed, an operator action time of 3 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions were performed at 2512 seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 6.3.1-2.

   The intact steam generator's PORV, which was used for the cooldown, also automatically open (at about 2550 seconds) to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORV is opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

6.3.1.4 Acceptance Criteria

The analysis is performed to demonstrate that the secondary side of the ruptured steam generator does not completely fill with water. The available secondary side volume of a single HNP Model Delta 75 replacement steam generator is 5545 ft^3. Margin to overfill is demonstrated provided the transient calculated steam generator secondary side water volume is less than 5545 ft^3.

6.3.1.5 Results

The primary to secondary break flow rate throughout the recovery operations is presented in Figure 6.3.1-5. The water volume in the ruptured steam generator is presented as a function of time in Figure 6.3.1-6. The secondary side volume of a single HNP Model Delta 75 replacement steam generator, up to the outlet nozzle, is 5545 ft^3. The peak ruptured steam generator water volume of 5285 ft^3 is indicated in Figure 6.3.1-6 showing that there is 260 ft^3 of margin to overfill. No credit is taken for the volume of the nozzle or any steam piping. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for HNP with the replacement steam generators and operation at the uprated NSSS power of 2912.4 MWt.

The difference in plant operating parameters and initial conditions between the uprated and current NSSS power levels has a small impact on the margin to overfill analysis. This has been confirmed via sensitivity runs made specifically for the HNP SGR/Uprating program. The analysis performed at the uprated power level demonstrated significant margin to overfill. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for HNP with the replacement steam generators and operation at the current NSSS power of 2787.4 MWt.

6.3.1.6 Conclusions

It is calculated that overfill of the ruptured steam generator will not occur for a design basis SGTR for HNP with the replacement steam generators and operation at either the uprated (2912.4 MWt) or the current (2787.4 MWt) NSSS power.

6.3.1.7 References

1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Lewis, Huang, Behnke, Fittante, Gelman, August 1987.

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

3. WCAP-12403, "LOFTTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.

4. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz , November 1992.

Table 6.3.1-1 Operator Action Times For Design Basis SGTR Analysis		
Action	**Time**	**Reference 3 and 4 Time**
Isolate auxiliary feedwater flow to ruptured SG	Maximum of 10 min. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG	Maximum of 13 min. 35 sec. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Isolate steam flow from ruptured SG	Maximum of 12 min. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG	Maximum of 13 min. 35 sec. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Operator action time to initiate cooldown	5 min. from time of steamline isolation*	8 min. 1 sec. from time of steamline isolation
Cooldown	Calculated by LOFTTR2	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 minutes from end of cooldown	2 min. 16 sec. from end of cooldown
Depressurization	Calculated by LOFTTR2	Calculated by LOFTTR2
Operator action time to terminate SI following depressurization	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria
Pressure equalization	Calculated by LOFTTR2	Calculated by LOFTTR2

* The analysis considered the possibility that the MSIVs would close as a result of the loss of offsite power assumed to occur coincident with reactor trip by isolating the steam lines at that time. The steamline isolation step was retained to model the time when the operators reach the step in the EOPs requiring that isolation of all flow in and out of the ruptured steam generator is isolated.

Table 6.3.1-2 Sequence of Events for Margin to Overfill Analysis	
	Operation at 2912.4 MWt
Event	**Time (seconds)**
Steam Generator Tube Rupture	0
Reactor Trip	113
SI Actuation	183
AFW Flow to Ruptured SG Isolated	600
Ruptured SG Isolated	720
RCS Cooldown Initiated	1020
RCS Cooldown Terminated	1960
RCS Depressurization Initiated	2200
RCS Depressurization Terminated	2332
SI Terminated	2512
Break Flow Terminated	3024

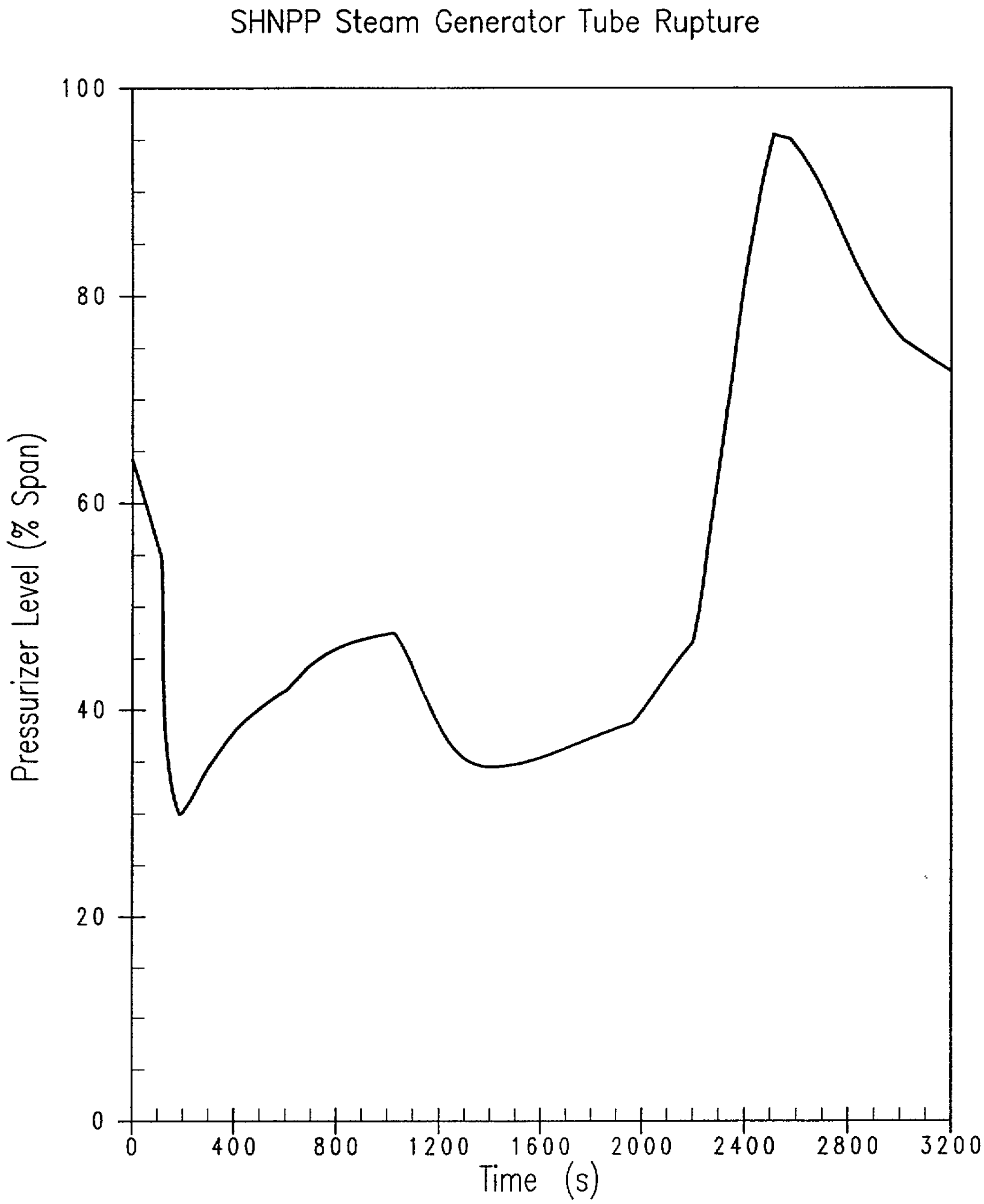

Figure 6.3.1-1
Pressurizer Level –
Margin to Overfill Analysis (NSSS Power of 2912.4 MWt)

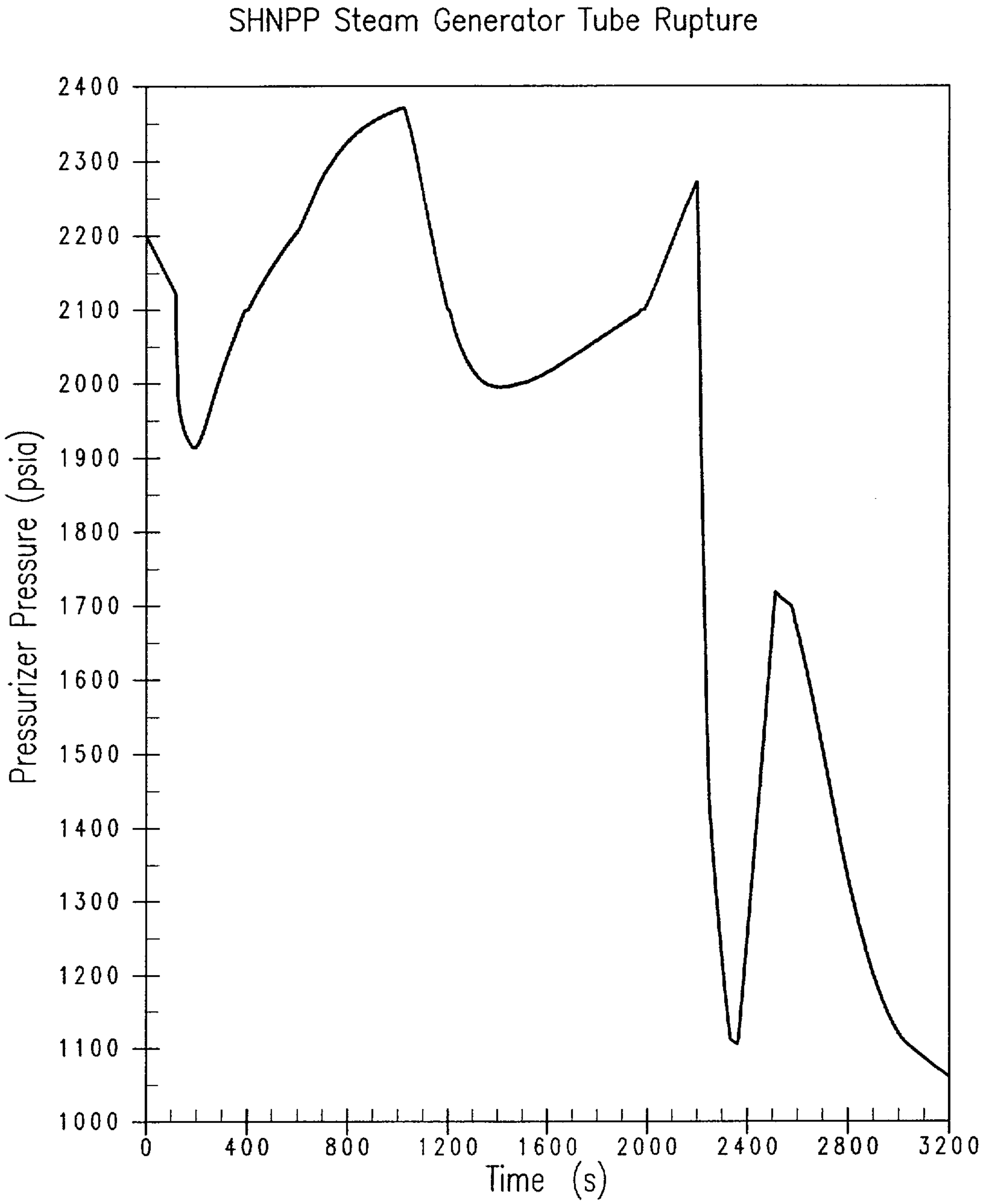

Figure 6.3.1-2
Pressurizer Pressure –
Margin to Overfill Analysis (NSSS Power of 2912.4 MWt)

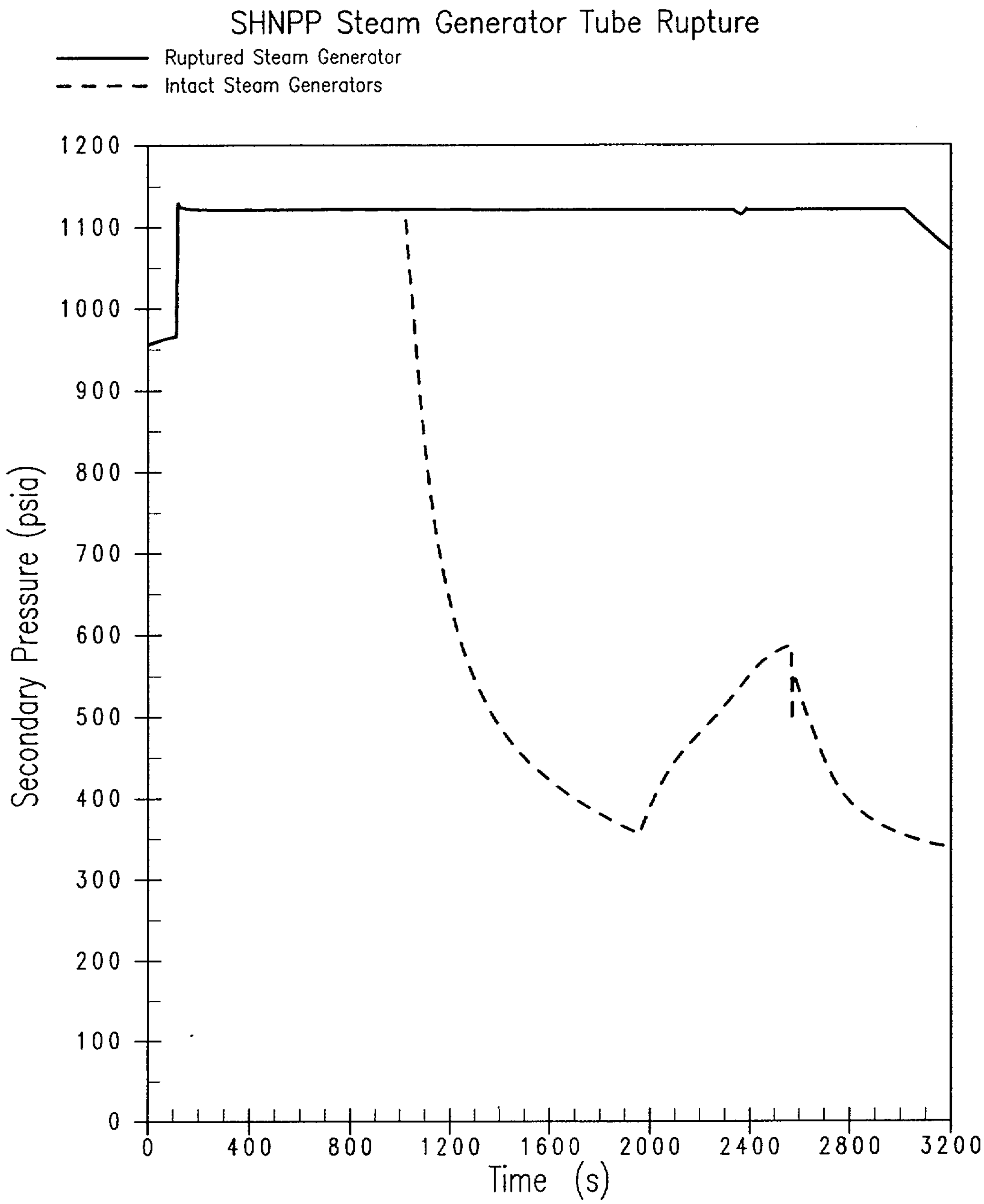

Figure 6.3.1-3
Secondary Pressure –
Margin to Overfill Analysis (NSSS Power of 2912.4 MWt)

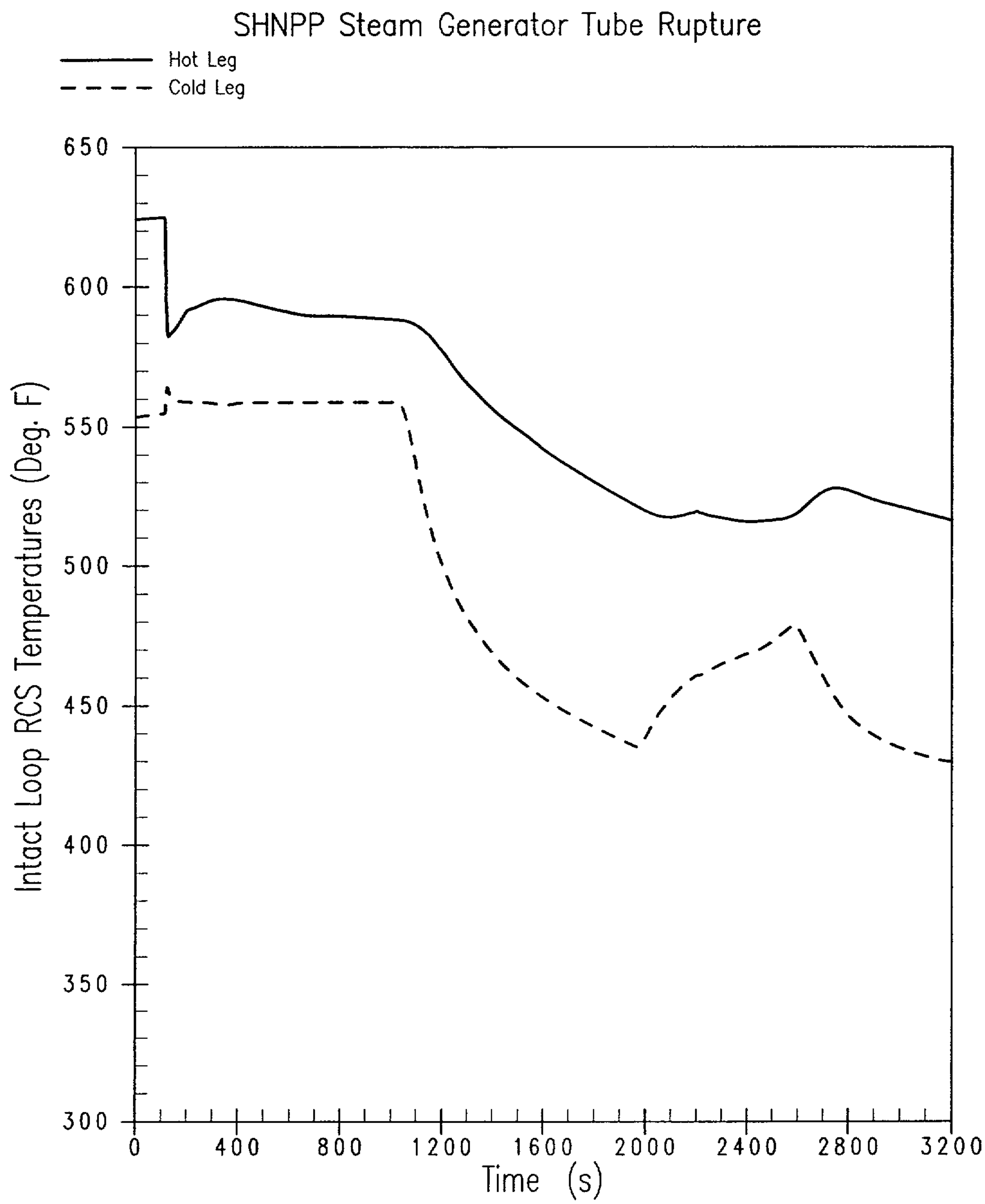

Figure 6.3.1-4
Intact Loop Hot & Cold LegTemperatures –
Margin to Overfill Analysis (NSSS Power of 2912.4 MWt)

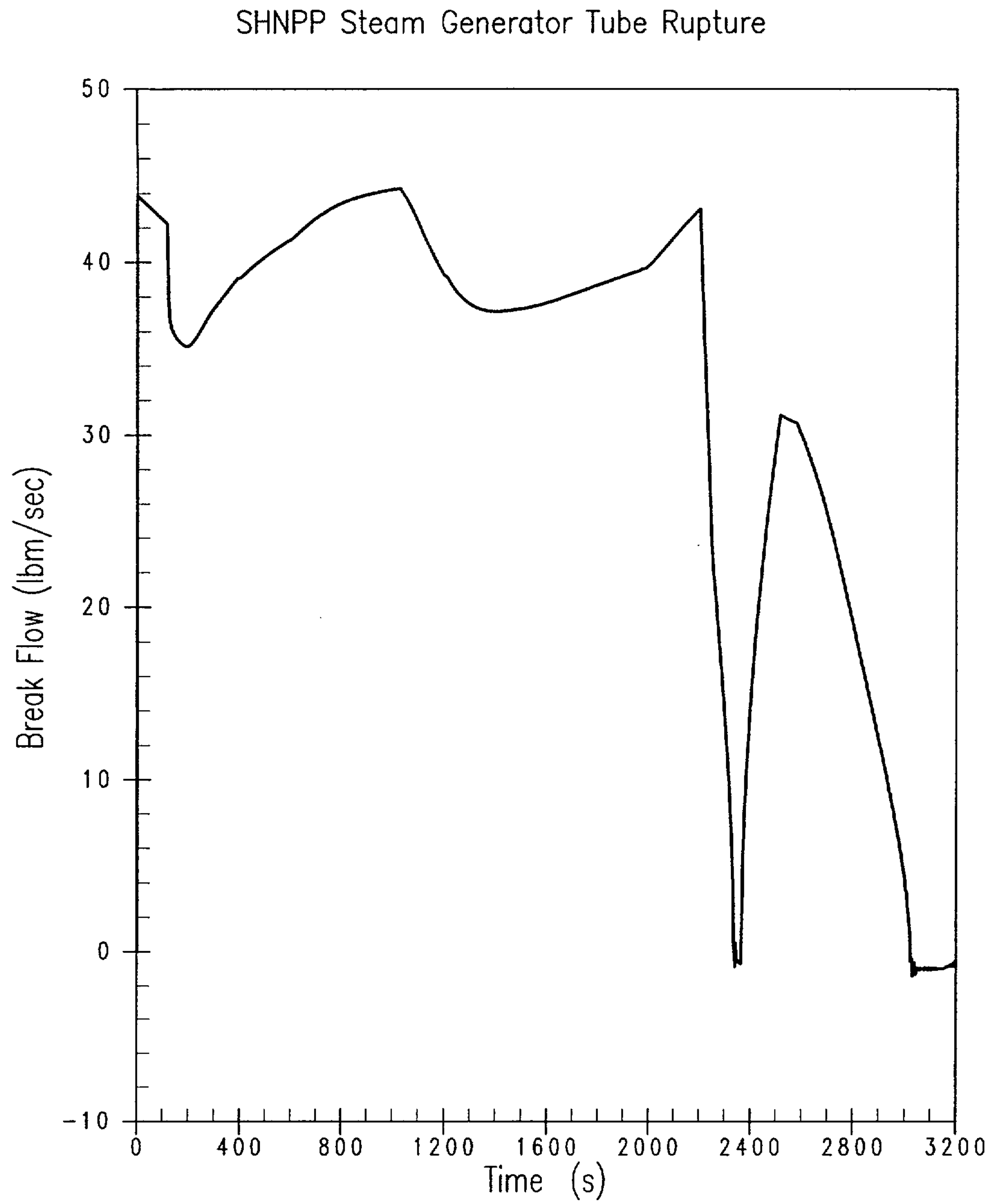

Figure 6.3.1-5
Primary to Secondary Break Flow –
Margin to Overfill Analysis (NSSS Power of 2912.4 MWt)

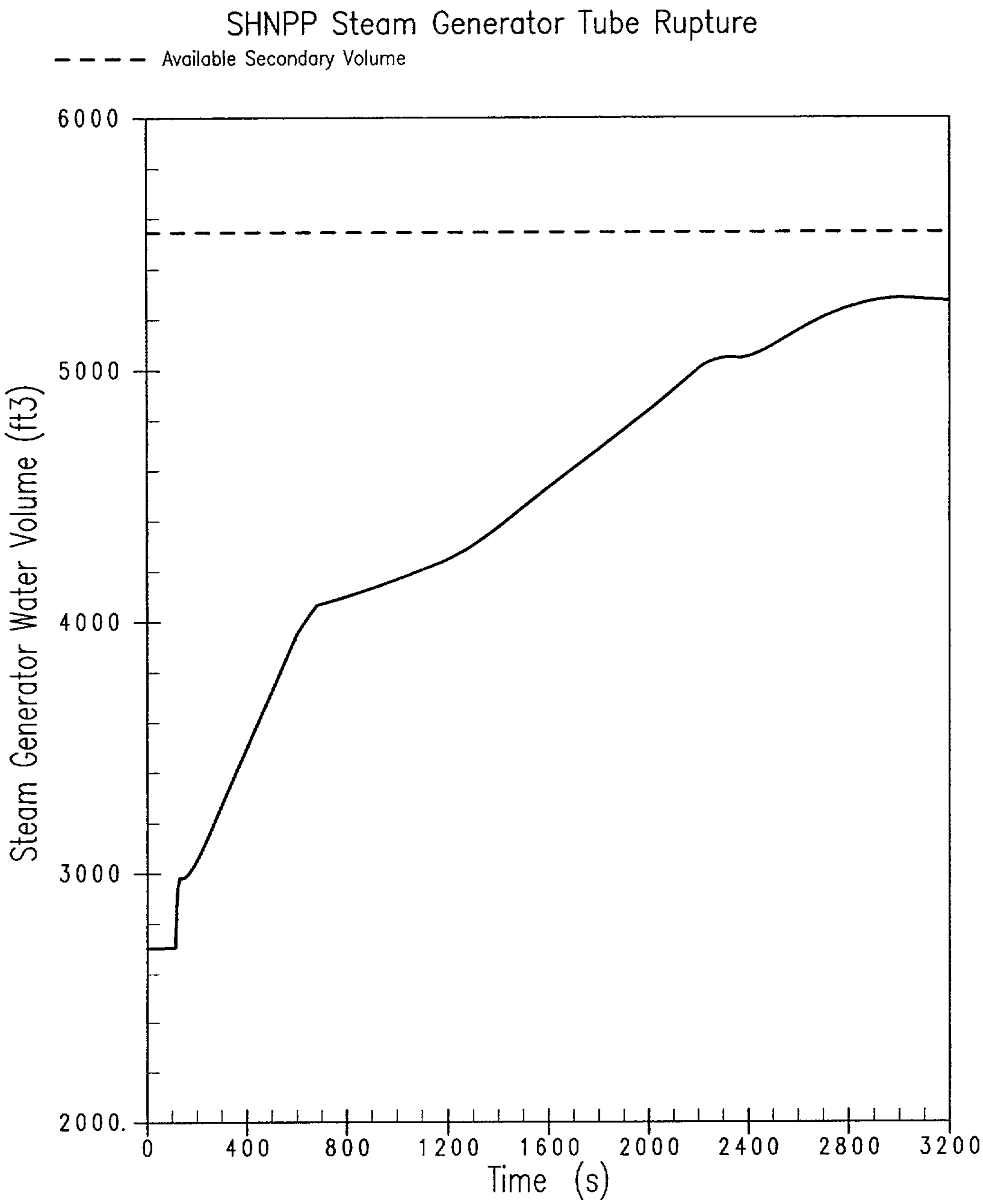

Figure 6.3.1-6
Ruptured SG Water Volume –
Margin to Overfill Analysis (NSSS Power of 2912.4 MWt)

6.3.2 Thermal and Hydraulic Analysis for Offsite Radiological Consequences

6.3.2.1 Introduction

The Steam Generator Tube Rupture (SGTR) analyses were performed for HNP using the analysis methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in Safety Evaluation Reports (SERs) dated December 17, 1985 and March 30, 1987. The methodology was developed for use with the LOFTTR2 program, an updated version of the LOFTTR1 program. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in References 1 and 2. This is the same methodology employed in the most recent analyses performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 3 and 4).

A thermal and hydraulic analysis was performed to determine the input for the offsite radiological consequences analysis for a design basis SGTR event for HNP. The thermal and hydraulic analysis was performed using the LOFTTR2 program and the methodology developed in References 1 and 2, and using the plant specific parameters for HNP. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

6.3.2.2 Description of Analyses and Evaluations

The thermal-hydraulic analysis, which determines the offsite dose mass releases, assumes that the plant is operating with the feedwater temperature at the lower end of the temperature window, since this was determined to result in slightly higher releases. No tube plugging is assumed in the analysis as this maximizes heat transfer to the ruptured steam generator. A high heat transfer rate during the transient maximizes the amount of mass released from the steam generator due to steaming. Although maximum tube plugging results in a lower initial water mass in the ruptured steam generator, which leads to increased steam releases, this is not more limiting than the higher heat transfer corresponding to no tube plugging. This has been confirmed via sensitivity runs specifically made for the HNP replacement steam generator and uprate program.

Design Basis Accident

The design basis accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold-leg) side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary to secondary leakage than a break on the hot side of the steam generator, as determined by Reference 1. However, as indicated subsequently, the break flow flashing fraction was conservatively calculated assuming that all of the break flow comes from the hot-leg side of the steam generator. The combination of these conservative assumptions regarding the break location results in a very conservative calculation of the offsite radiation doses. It was also assumed that loss-of-offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. Due to the assumed loss of offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator PORVs. After reactor trip and loss of offsite power, the RCPs begin to coast down.

Based on the information in Reference 2, the most limiting single failure with respect to offsite doses is a failed open power-operated relief valve (PORV) on the steam generator with the ruptured tube. Failure of this PORV will cause an uncontrolled depressurization of the steam generator, which will increase primary to secondary leakage and the mass release to the atmosphere. Pressure in the ruptured steam generator will remain below that in the primary system until the failed PORV can be isolated, and recovery actions completed.

Conservative Assumptions

The integrated primary-to-secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere until break flow termination were calculated with the LOFTTR2 program. This is used in calculating the offsite radiation doses. This section includes a discussion of the methods and assumptions used to analyze the SGTR event and to calculate the mass releases, the sequence of events during the recovery operations, and the calculated results.

Most of the conservative conditions and assumptions used for the margin to overfill analysis are also conservative for the offsite dose analysis, and thus most of the same assumptions were used for both analyses. The major differences in the assumptions that were used for the LOFTTR2 analysis for offsite doses are discussed below.

1. Reactor Trip and Turbine Runback

   An earlier reactor trip is conservative for the offsite dose analysis. Due to the assumed loss of offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator PORVs. Thus an earlier trip time leads to more steam released to the atmosphere from the ruptured and intact steam generators. The time of reactor trip was calculated by modeling the HNP protection system and this time was used in the analysis. However, turbine runback was not simulated for the analysis since the increased secondary water mass due to turbine runback is not conservative for the offsite dose analysis.

2. Steam Generator Secondary Mass

   A lower initial mass in the ruptured steam generator results in a conservative prediction of offsite doses. The initial steam generator total fluid mass was assumed to be 10-percent below the nominal full-power fluid mass.

3. Auxiliary Feedwater (AFW) System Operation

   For this analysis, the minimum AFW flow rate of 390 gpm to the ruptured steam generator was assumed to be initiated 61.5 seconds after reactor trip. A minimum AFW flow rate maximizes steam releases to the atmosphere.

4. Flashing Fraction

   When calculating the fraction of break flow that flashes to steam, 100 percent of the break flow is assumed to come from the hot-leg side of the break. Since the tube rupture flow actually consists of flow from the hot-leg and cold-leg sides of the steam generator, the temperature of the combined flow will be less than the hot-leg temperature and the flashing fraction will be correspondingly lower. Thus the assumption is conservative for a SGTR analysis.

Operator Action Times

The major operator actions required for the recovery from a SGTR are discussed in Section 6.3.1.2, and the operator action times used for the margin to overfill analysis are presented in Table 6.3.1-1. The operator action times assumed for the margin to overfill analysis were also used for the offsite dose analysis. However, for the offsite doses analysis, the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator is isolated. Before proceeding with the recovery operations, the failed-open PORV on the ruptured steam generator is assumed to be isolated by locally closing the associated block valve. CP&L has determined that an operator can locally close the block valve for the PORV on the ruptured steam generator within 20 minutes after the failure. Thus, it was assumed that the ruptured steam generator PORV is isolated at 20 minutes after the valve is assumed to fail open. The operator action time to close the block valve for the PORV on the ruptured steam generator is the same as that modeled in the Reference 3 and 4 analyses. After the ruptured steam generator PORV is isolated, the additional delay time of 5 minutes (Table 6.3.1-1) was assumed for the operator action time to initiate the RCS cooldown.

Mass Releases

The mass releases were determined for use in evaluating the offsite and control room radiological consequences of the SGTR using the methodology of Reference 2. The steam releases from the ruptured and intact steam generators, the feedwater flow to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0 to 2 hours are used to calculate the radiation doses at the site boundary for a 2-hour exposure, and the releases for 0 to 8 hours are used to calculate the radiation doses at the low population zone and to the operators in the control room for the duration of the accident.

In the LOFTTR2 analyses, the SGTR recovery actions in the E-3 guideline were simulated until the termination of primary-to-secondary leakage. After the primary-to-secondary leakage is terminated, the operators will continue the SGTR recovery actions to prepare the plant for cooldown to cold shutdown conditions. When these recovery actions are completed, the plant should be cooled and depressurized to cold shutdown conditions. In accordance with the methodology in Reference 2 it was assumed that the cooldown is performed using HNP Emergency Operating Procedures (EOPs) ES-3.3, Post-SGTR Cooldown Using Steam Dump, since this method results in a conservative evaluation of the long-term releases for the offsite dose analysis compared to the other cooldown methods in the EOPs. This procedure for depressurizing the ruptured steam generator is assumed even though the LOFTTR2 analysis performed to calculate releases up until break flow termination has assumed PORV isolation.

The high level actions for the post-SGTR cooldown method using steam dump in the HNP EOP ES-3.3 are discussed below.

5. Prepare for Cooldown to Cold Shutdown.

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

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

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

7. Depressurize RCS to RHR System Pressure.

   When the cooldown to RHR system temperature is completed, the pressure in the ruptured steam generator is decreased by releasing steam from the ruptured steam generator. Steam release to the condenser is preferred, since this minimizes radiological releases, but steam can be released to the atmosphere using the PORV on the ruptured steam generator if the condenser is not available. Consistent with the assumption of a loss-of-offsite power, it was assumed that the ruptured steam generator is depressurized by releasing steam via the PORV. As the ruptured steam generator pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured steam generator in order to prevent in-leakage of secondary side water or additional primary to secondary leakage. Although normal pressurizer spray is the preferred means of RCS pressure control, auxiliary spray or pressurizer PORV can be used to control RCS pressure if pressurizer spray is not available.

8. Cooldown to Cold Shutdown.

   When RCS temperature and pressure have been reduced to the RHR system in-service values, RHR system cooling is initiated to complete the cooldown to cold shutdown. When cold shutdown conditions are achieved, the pressurizer can be cooled to terminate the event.

6.3.2.3 Description of Analysis Cases

The LOFTTR2 analysis results for the HNP offsite dose evaluation are described below, considering operation at the uprated NSSS power of 2912.4 MWt with Model Delta 75 replacement steam generators. The sequence of events for the analysis is presented in Table 6.3.2-1. The transient results for this case are similar to the transient results for the overfill analysis until the ruptured steam generator is isolated. The transient behavior is different after this time, as it is assumed that the ruptured steam generator PORV fails open at that time.

Following the tube rupture, the RCS pressure decreases as shown in Figure 6.3.2-1 due to the primary to secondary leakage. In response to this depressurization, the reactor trips on overtemperature-ΔT at about 114 seconds. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. The steam dump system is inoperable due to the assumed loss-of-offsite power, which results in the secondary pressure rising to the steam generator PORV setpoint as shown in Figure 6.3.2-2. The RCS pressure and pressurizer level also decrease more rapidly following reactor trip as shown in Figures 6.3.2-1 and 6.3.2-3. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at approximately 178 seconds.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator.

   Recovery actions begin by throttling the auxiliary feedwater flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously, isolation of the AFW flow to the ruptured steam generator was assumed to be completed at 10 minutes after the initiation of the SGTR or when the narrow range level reaches 30 percent, whichever time is greater. For the HNP analysis, the time to reach 30 percent is less than 10 minutes, and thus the ruptured steam generator is assumed to be isolated at 10 minutes. Also, as indicated previously, complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30 percent on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer. For the HNP analysis, the time to reach 30 percent is less than 12 minutes, and thus the ruptured steam generator is assumed to be isolated at 12 minutes.

The ruptured steam generator PORV is also assumed to fail open at this time. The failure causes the steam generator to rapidly depressurize, which results in an increase in primary to secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary, which results in RCS pressure and temperature decreasing more rapidly than in the margin to overfill analysis. The ruptured steam generator depressurization causes a cooldown in the intact steam generators loops. It is assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to locally close the associated block valve is 20 minutes. At 1922 seconds the depressurization of the ruptured steam generator is terminated and the ruptured steam generator pressure begins to increase as shown in Figure 6.3.2-2.

2. Cooldown the RCS to establish Subcooling Margin.

After the block valve for the ruptured steam generator PORV is closed, there is a 5 minute operator action time imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator due to the failed-open PORV affects the RCS cooldown target temperature since the temperature is determined based upon the pressure in the ruptured steam generator at that time. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact steam generators' PORVs. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 20°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator when the cooldown is initiated (relative to the margin to overfill analysis), the associated temperature to which the RCS must be cooled is also lower. However, both intact steam generator PORVs are available for the cooldown in this case, whereas in the margin to overfill analysis the limiting single failure resulted in only one intact steam generator being available for the cooldown. The cooldown begins at 2224 seconds and is completed at 2996 seconds.

The reduction in the intact steam generators' pressure required to accomplish the cooldown is shown in Figure 6.3.2-2 and the effect of the cooldown on the RCS temperatures is shown in Figures 6.3.2-4 and 6.3.2-5. The RCS pressure and pressurizer level also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 6.3.2-1 and 6.3.2-3. The break flow flashing fraction is calculated throughout the transient based on the difference between the enthalpy of the break flow and the saturation enthalpy at the ruptured steam generator pressure as shown in Figure 6.3.2-7. Break flow is calculated to stop flashing at approximately 2500 seconds as a result of the reduction in primary coolant temperature associated with the cooldown (Figure 6.3.2-4) and the increase in ruptured steam generator pressure following isolation of the failed open PORV (Figure 6.3.2-2).

3. Depressurize to Restore Inventory.

After the RCS cooldown is completed, a 240 second operator action time is included prior to the RCS depressurization. The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The RCS depressurization is initiated at 3236 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 10 percent for pressurizer level uncertainty, or pressurizer level is greater than 75 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated at 3312 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10 percent. The RCS depressurization reduces the break flow as shown in Figure 6.3.2-6 and increases SI flow to refill the pressurizer, as shown in Figure 6.3.2-3.

4. Terminate SI to Stop Primary to Secondary Leakage.

   The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 20°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 10 percent allowance for uncertainty.

   After depressurization is completed, an operator action time of 3 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions were performed at 3492 seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 6.3.2-1.

   The intact steam generators' PORVs also automatically open to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORVs are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The ruptured steam generator pressure increases to the PORV setpoint and steam release is reinitiated. Steam generator pressure is maintained at the steam generator PORV setpoint rather than the safety valve setpoint for modeling efficiency. This modeling is conservative since it delays break flow termination by requiring the RCS pressure to drop further, maximizes the break flow rate by maintaining a larger primary-to-secondary pressure differential, and results in more steam release from the ruptured steam generator. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

Calculation of Mass Releases

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage are simulated in the LOFTTR2 analyses. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, it was assumed that the RCS and intact steam generators conditions are maintained stable for a 20 minute period until the cooldown to cold shutdown is initiated. The PORVs for the intact steam generators were then assumed to be used to cool down the RCS to the RHR system operating temperature of 325°F, at the maximum allowable cooldown rate of 100°F/hr. The RCS and the intact steam generators temperatures at 2 hours were then determined using the RCS and intact steam generators parameters at the time of leakage termination and the RCS cooldown rate. The steam releases and the feedwater flows for the intact steam generators for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact steam generators conditions at the time of leakage termination and at 2 hours. Since the ruptured steam generator is isolated, no change in the ruptured steam generator conditions is assumed to occur until subsequent depressurization.

The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 325°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed immediately following the completion of the RCS cooldown. The ruptured steam generator was assumed to be depressurized to the RHR in-service pressure of 365 psia via steam release from the ruptured steam generator PORV, since this maximizes the steam release from ruptured steam generator to the atmosphere which is conservative for the evaluation of the offsite radiation doses. The RCS pressure is also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It is assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact steam generators from a mass and energy balance using the RCS and steam generator conditions at 2 hours and at the RHR system in-service conditions. The steam released from the ruptured steam generator from 2 to 8 hours was determined based on a mass and energy balance for the ruptured steam generator using the conditions at the time of leakage termination and saturated conditions at the RHR in-service pressure.

After 8 hours, it is assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR in-service conditions are assumed to be reached at 8 hours.

The analysis was also run at the current NSSS power level of 2787.4 MWt, since the assumed power level impacts the calculated offsite and control room doses.

6.3.2.4 Acceptance Criteria

The analysis is performed to calculate the mass transfer data for input to the radiological consequences analysis. As such no acceptance criteria are defined. The results of the analysis are used as input to the radiological consequences analysis presented in Section 6.3.3.

6.3.2.5 Results

LOFTTR2 Analysis Results

The primary to secondary break flow rate throughout the recovery operations is presented in Figure 6.3.2-6. The calculated break flow flashing fraction and integrated flashed break flow are presented in Figures 6.3.2-7 and 6.3.2-8, respectively. The ruptured steam generator PORV steam release rate is presented in Figure 6.3.2-9. The total intact steam generator PORV steam release rate is presented in Figure 6.3.2-10. The ruptured steam water volume is shown in Figure 6.3.2-11. For this case, the water volume in the ruptured steam generator when the break flow is terminated is less than the volume for the margin to overfill case and significantly less than the total steam generator volume of 5545 ft^3. The ruptured steam water mass is shown in Figure 6.3.2-12.

The results for the analysis performed with the replacement steam generators at the current NSSS power level of 2787.4 MWt progress in the same manner. The sequence of events for the analysis is included in Table 6.3.2-1. The primary to secondary break flow rate throughout the recovery operations is presented in Figure 6.3.2-13. The calculated break flow flashing fraction and integrated flashed break flow are presented in Figures 6.3.2-14 and 6.3.2-15, respectively. The ruptured steam generator PORV steam release rate is presented in Figure 6.3.2-16. The total intact steam generator PORV steam release rate is presented in Figure 6.3.2-17.

Mass Release Results

The mass release calculations were performed using the methodology discussed above. For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip will be through the condenser vacuum exhaust. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator PORVs.

The mass releases for the SGTR event assuming failure and isolation of the ruptured steam generator PORV are presented in Table 6.3.2-2, for the analysis modeling the Model Delta 75 replacement steam generators and the uprated NSSS power of 2912.4 MWt. The results indicate that approximately 138,300 lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours, 35,100 lbm of steam is released to the atmosphere from the ruptured steam generator. A total of 167,900 lbm of primary water is transferred to the secondary side of the ruptured steam generator before break flow is terminated. A total of 10,843 lbm of this break flow is assumed to flash to steam upon entering the steam generator.

The mass releases for the analysis modeling the Model Delta 75 replacement steam generators and the current NSSS power of 2787.4 MWt are presented in Table 6.3.2-3. The results indicate that approximately 135,700 lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours, 34,400 lbm of steam is released to the atmosphere from the ruptured steam generator. A total of 165,400 lbm of primary water is transferred to the secondary side of the ruptured steam generator before break flow is terminated. A total of 10,598 lbm of this break flow is assumed to flash to steam upon entering the steam generator.

6.3.2.6 Conclusions

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis has been completed and data tabulated for the limiting case. The results of the analysis are used as input to the radiological consequences analysis presented in Section 6.3.3.

6.3.2.7 References

1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Lewis, Huang, Behnke, Fittante, Gelman, August 1987.

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

3. WCAP-12403, "LOFTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.

4. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz, November 1992.

Table 6.3.2-1 Sequence of Events for Offsite Radiation Dose Analysis		
	Operation at 2912.4 MWt	**Operation at 2787.4 MWt**
Event	**Time (seconds)**	**Time (seconds)**
Steam Generator Tube Rupture	0	0
Reactor Trip	114	114
SI Actuation	178	170
AFW Flow to Ruptured SG Isolated	600	600
Ruptured SG Isolated	720	720
Ruptured SG PORV Fails Open	722	722
Ruptured SG Block Valve Closed	1922	1922
RCS Cooldown Initiated	2224	2224
RCS Cooldown Terminated	2996	2968
RCS Depressurization Initiated	3236	3208
RCS Depressurization Terminated	3312	3288
SI Terminated	3492	3468
Break Flow Terminated	4652	4468

Table 6.3.2-2 Mass Releases for Operation At NSSS Power of 2912.4 MWt; Total Mass Flow (Pounds)				
	Time Period			
	Time Zero to Time of Reactor Trip*	**Time of Reactor Trip to Time at Which Break Flow is Terminated***	**Time at Which Break Flow is Terminated to 2 Hours**	**2 Hours to Time at Which RCS Reaches RHR In-Service Conditions***
Ruptured SG				
- Condenser	128,300	0	0	0
- Atmosphere	0	138,300	0	35,100
- Feedwater	123,400	33,000	0	0
Intact SGs				
- Condenser	254,100	0	0	0
- Atmosphere	0	176,900	183,300	862,800
- Feedwater	254,100	292,400	201,800	894,900
Break Flow	4900	163,000	0	0
Flashed Break Flow	830	10,013	0	0

* Reactor trip occurs at 114 seconds; break flow is terminated at 4652 seconds; RHR conditions are reached at 8 hours.

Table 6.3.2-3 Mass Releases for Operation At NSSS Power of 2787.4 MWt; Total Mass Flow(Pounds)				
	Time Period			
	Time Zero to Time of Reactor Trip*	**Time of Reactor Trip to Time at Which Break Flow is Terminated***	**Time at Which Break Flow is Terminated to 2 Hours**	**2 Hours to Time at Which RCS Reaches RHR In-Service Conditions***
Ruptured SG				
- Condenser	122,900	0	0	0
- Atmosphere	0	135,700	0	34,400
- Feedwater	118,000	32,500	0	0
Intact SGs				
- Condenser	243,400	0	0	0
- Atmosphere	0	166,500	194,100	823,100
- Feedwater	243,400	281,000	215,000	850,600
Break Flow	4,800	160,600	0	0
Flashed Break Flow	794	9,804	0	0

* Reactor trip occurs at 114 seconds; break flow is terminated at 4468 seconds; RHR conditions are reached at 8 hours.

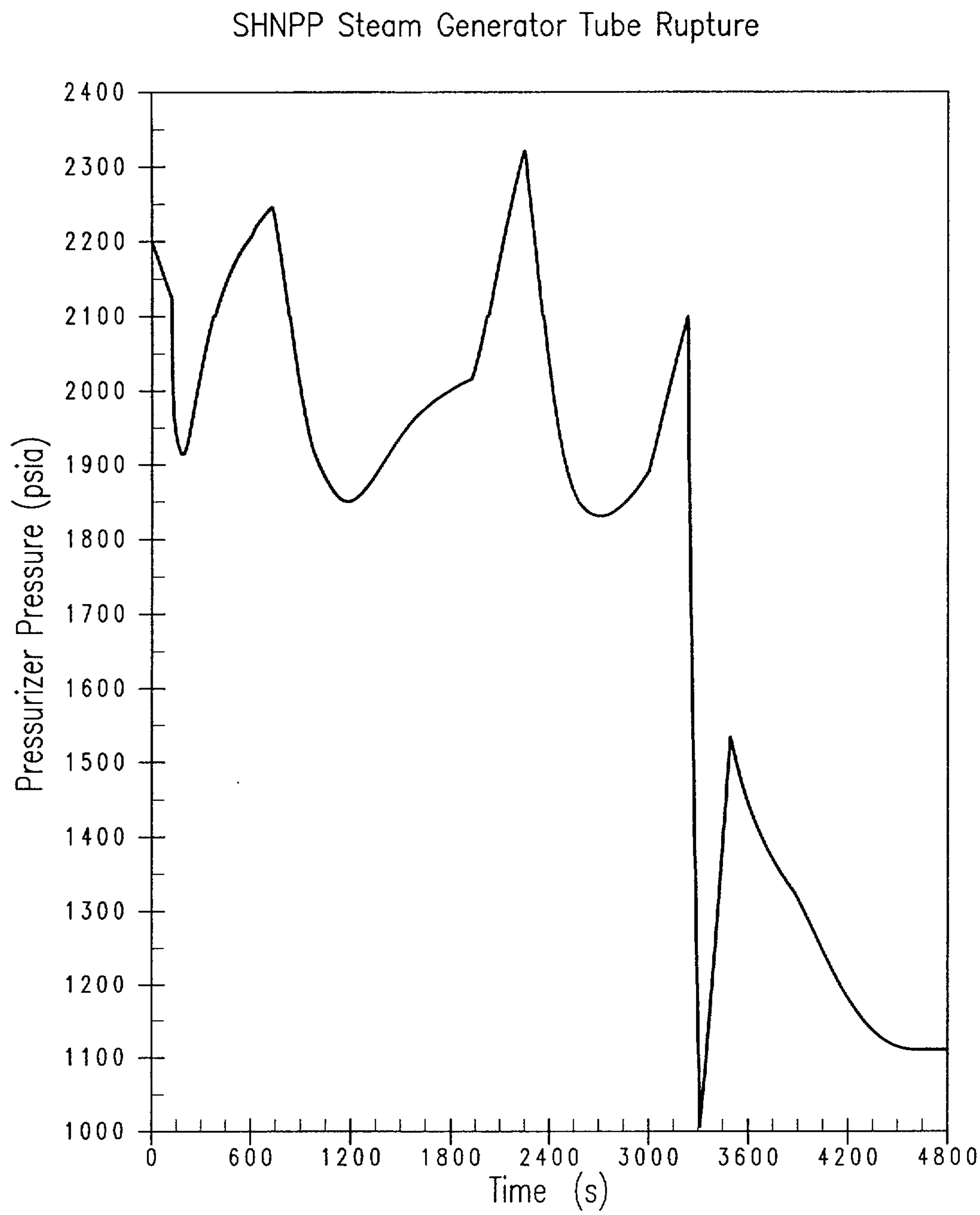

Figure 6.3.2-1
Pressurizer Pressure – Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

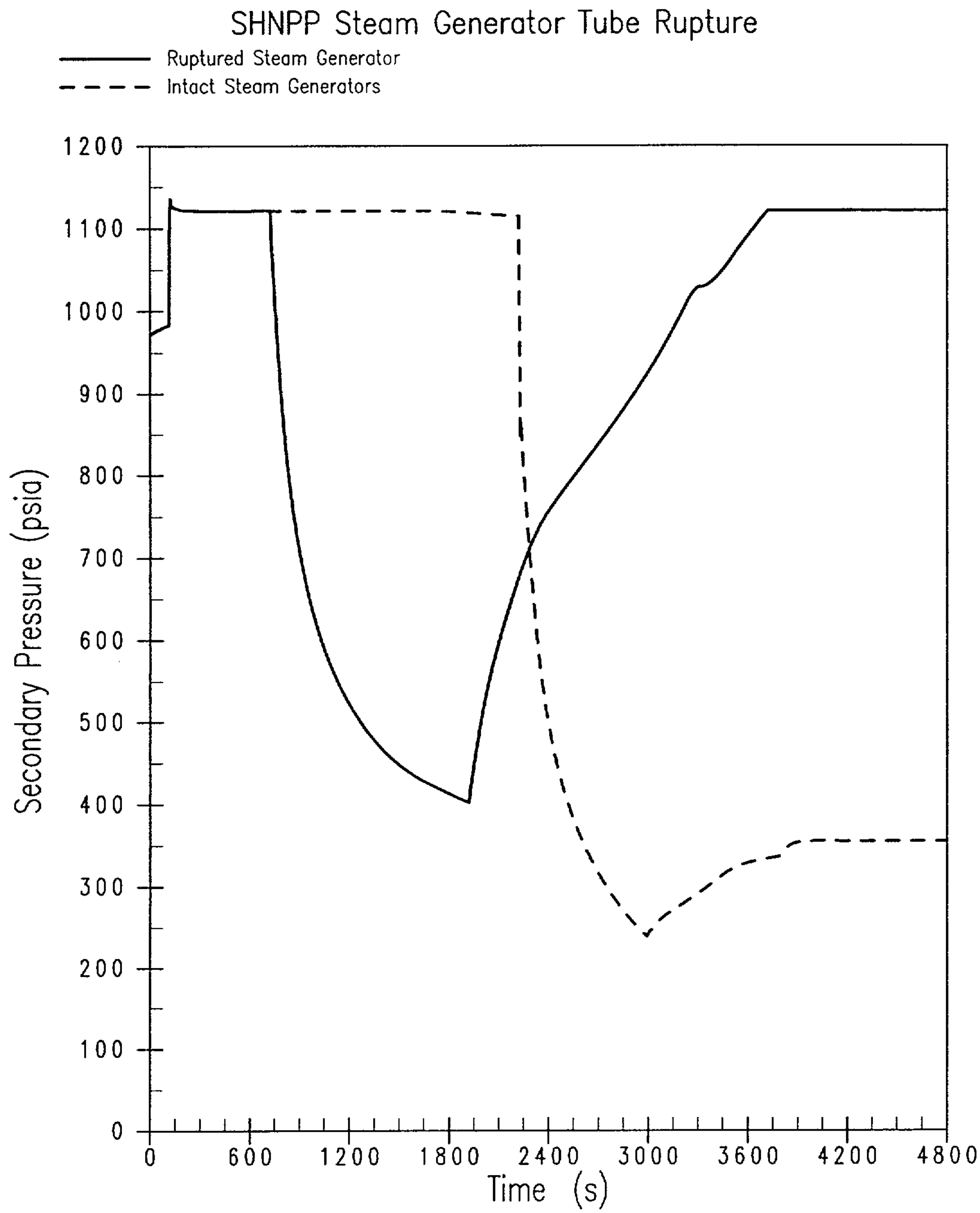

Figure 6.3.2-2
Secondary Pressure – Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

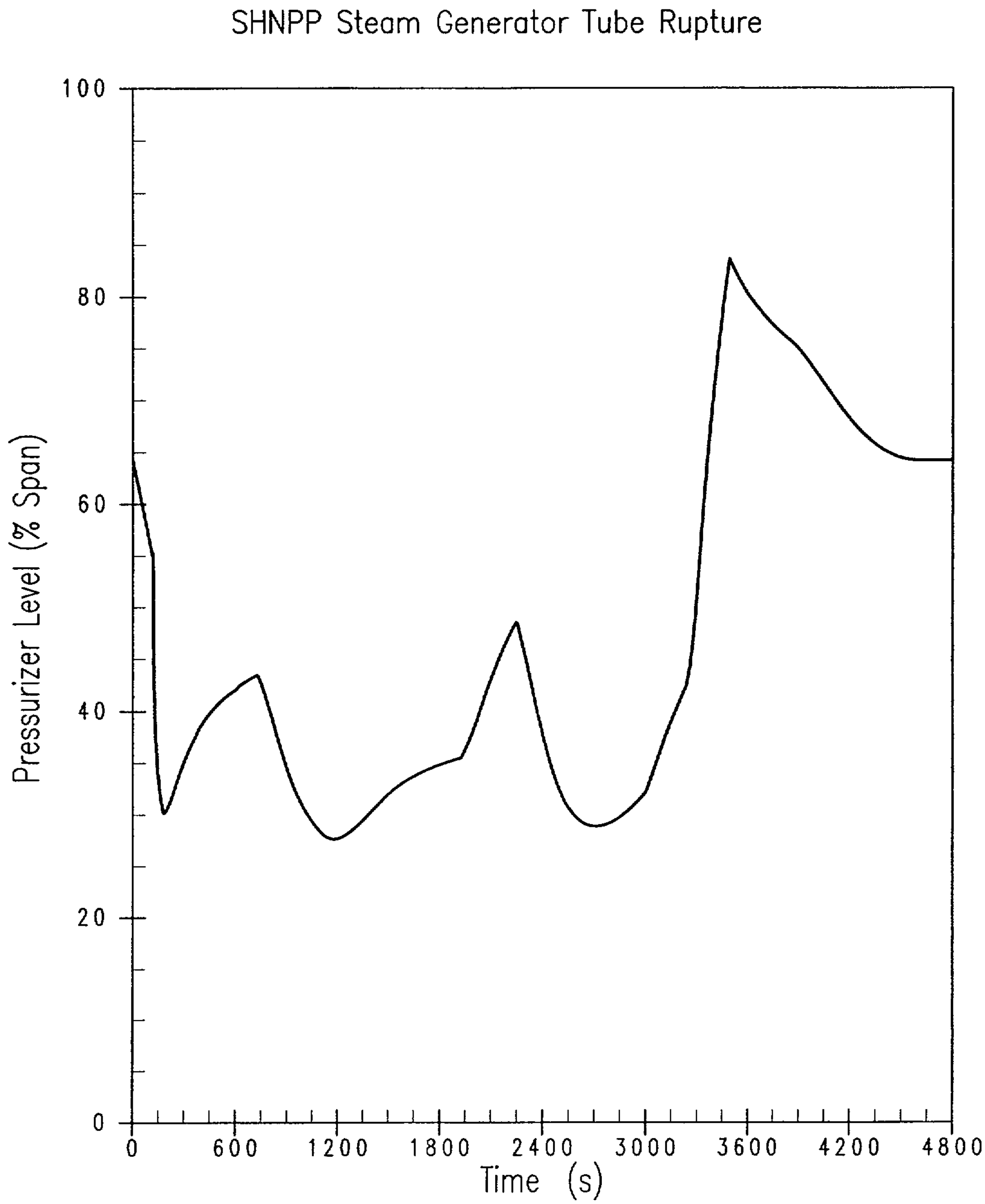

Figure 6.3.2-3
Pressurizer Level – Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

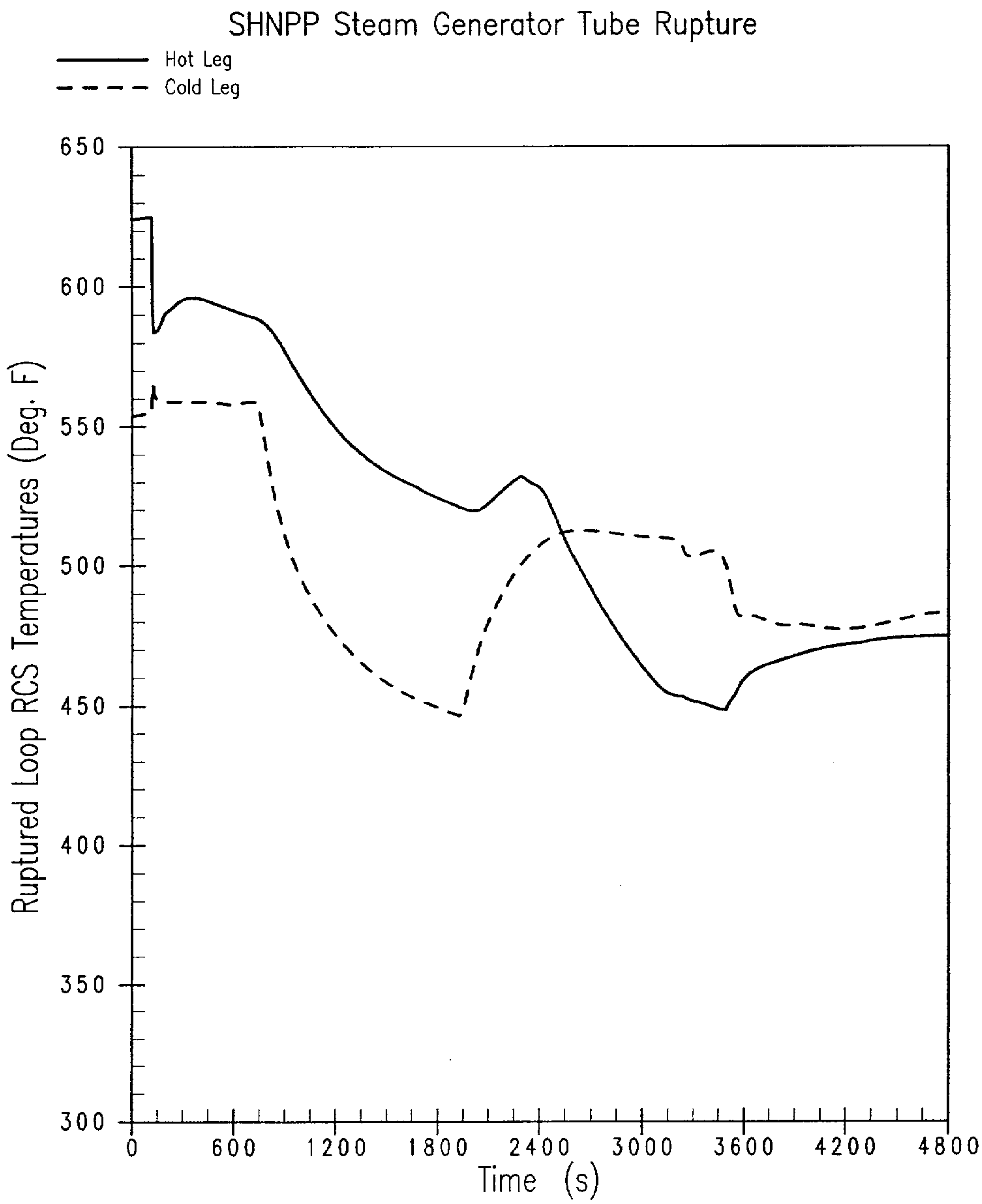

Figure 6.3.2-4
Ruptured Loop Hot & Cold LegTemperatures –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

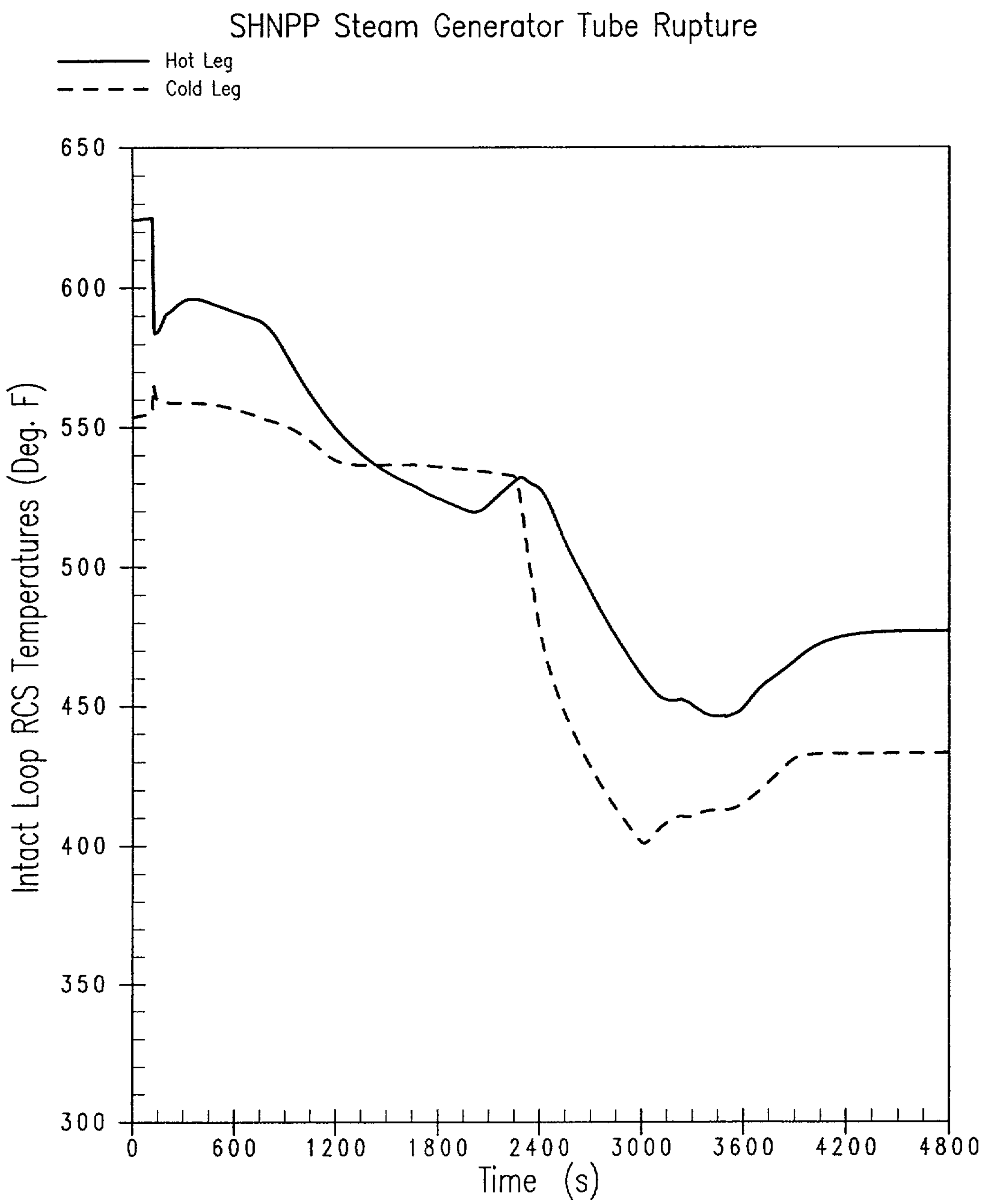

Figure 6.3.2-5
Intact Loop Hot & Cold LegTemperatures –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

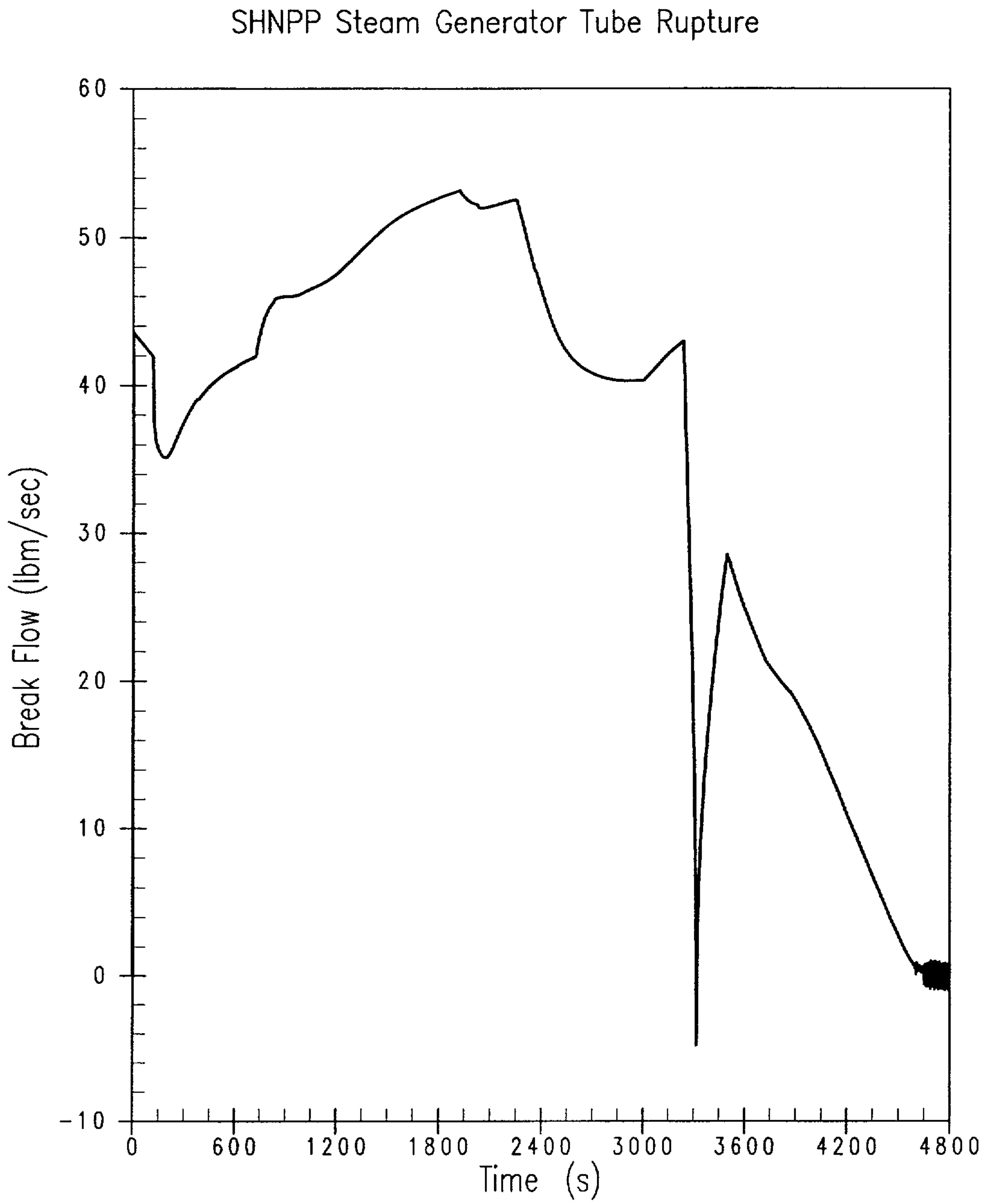

Figure 6.3.2-6
Primary to Secondary Break Flow –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

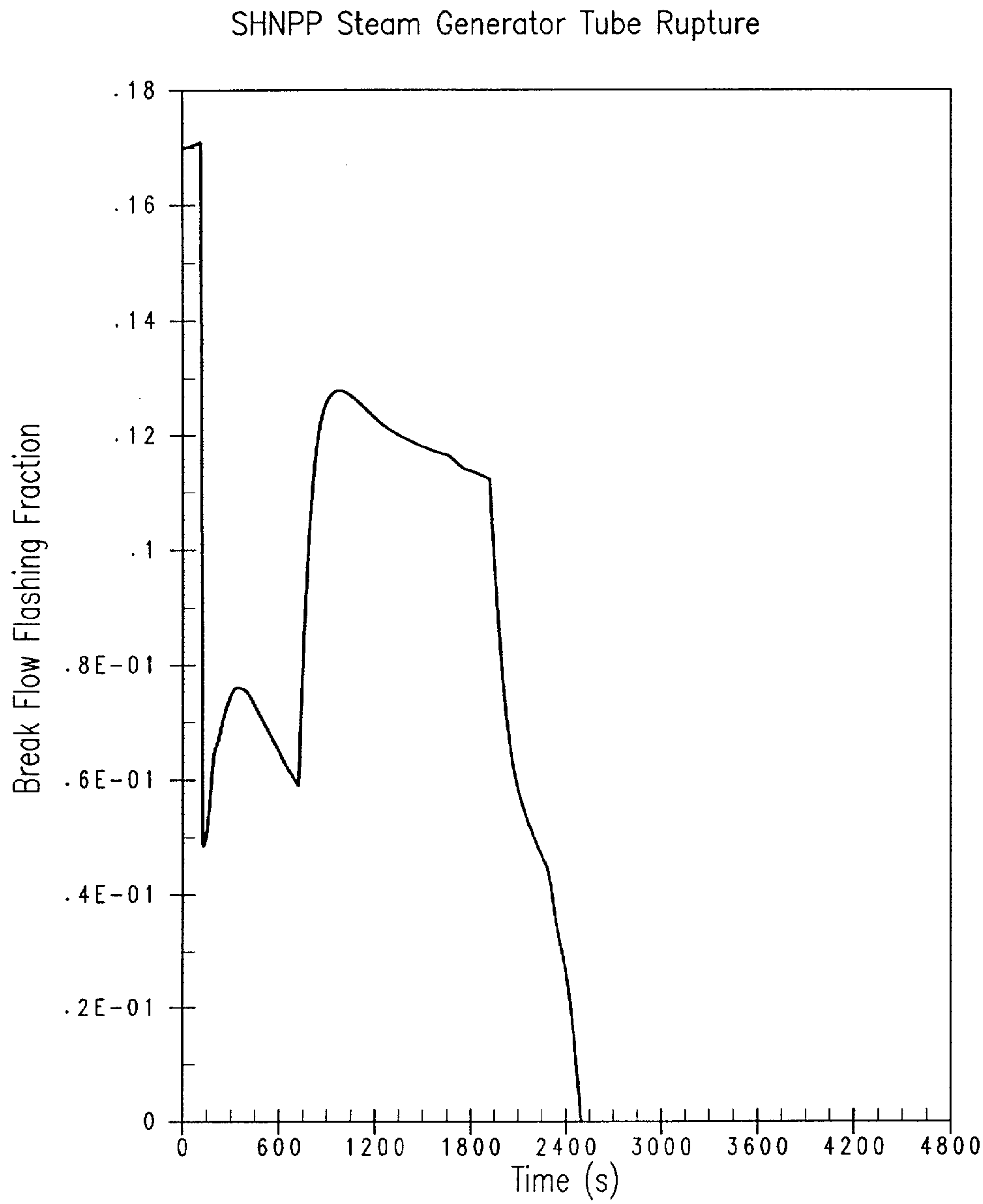

Figure 6.3.2-7
Break Flow Flashing Fraction –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

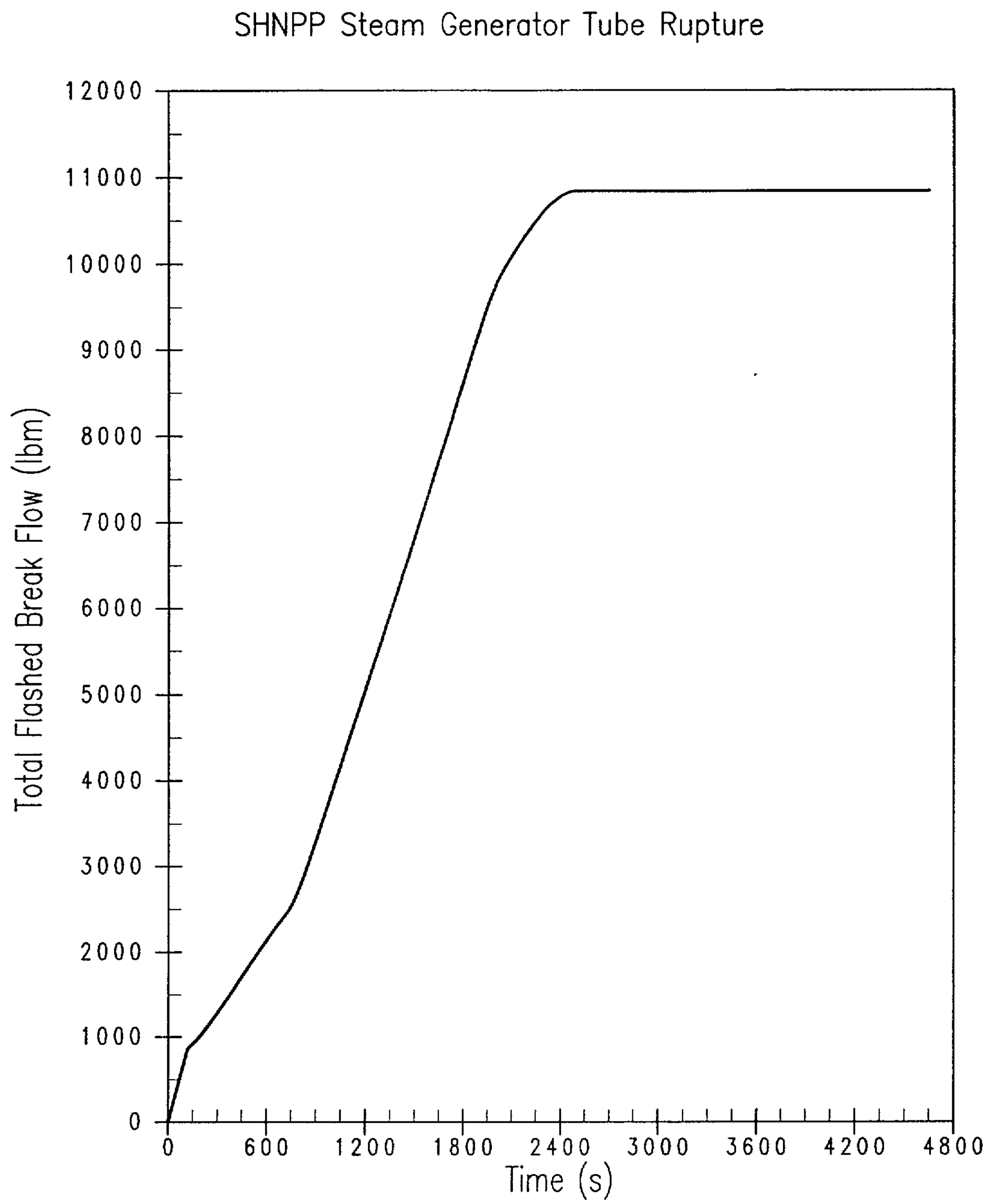

Figure 6.3.2-8
Total Flashed Break Flow –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

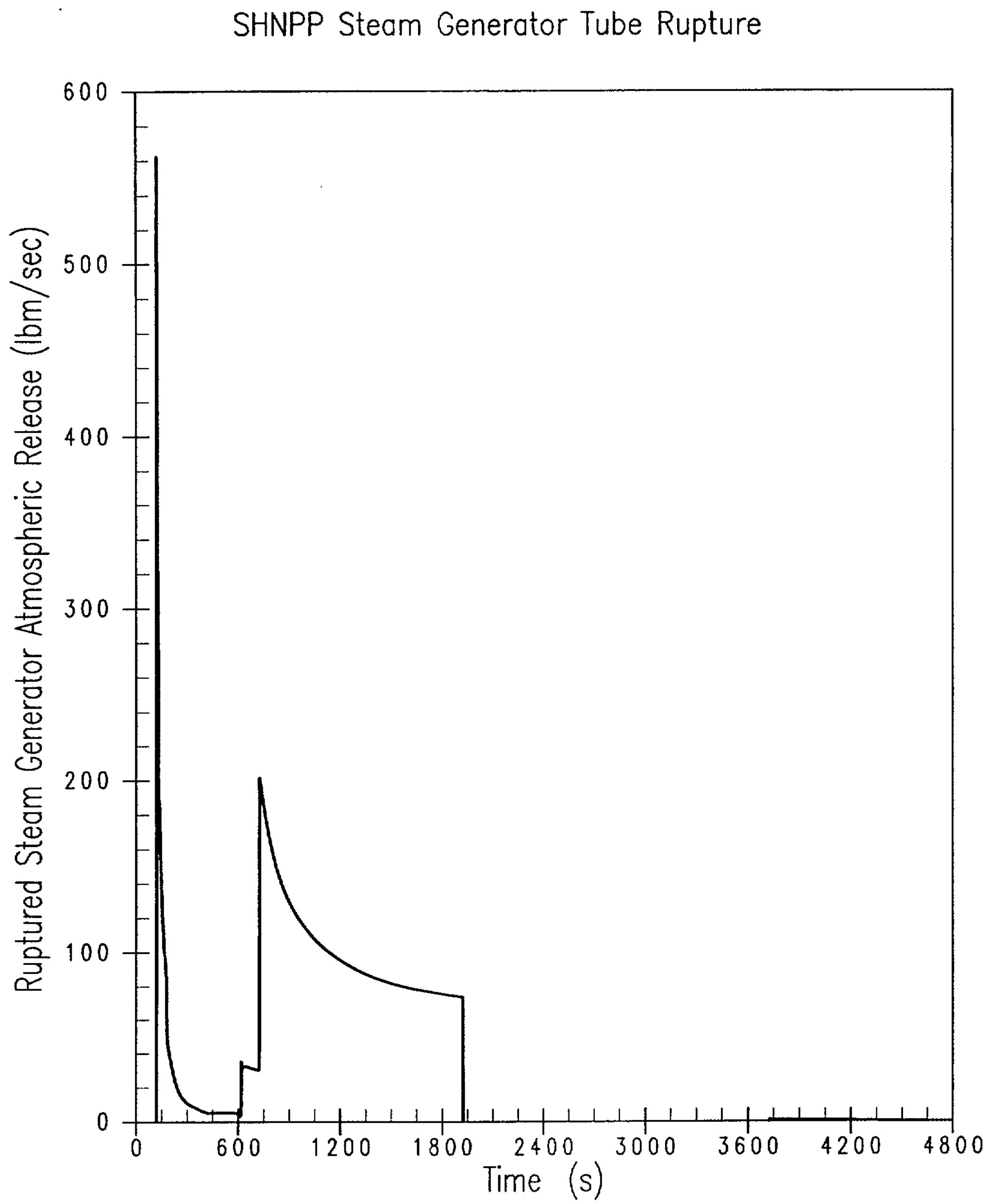

Figure 6.3.2-9
Ruptured SG Mass Release Rate to the Atmosphere – Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

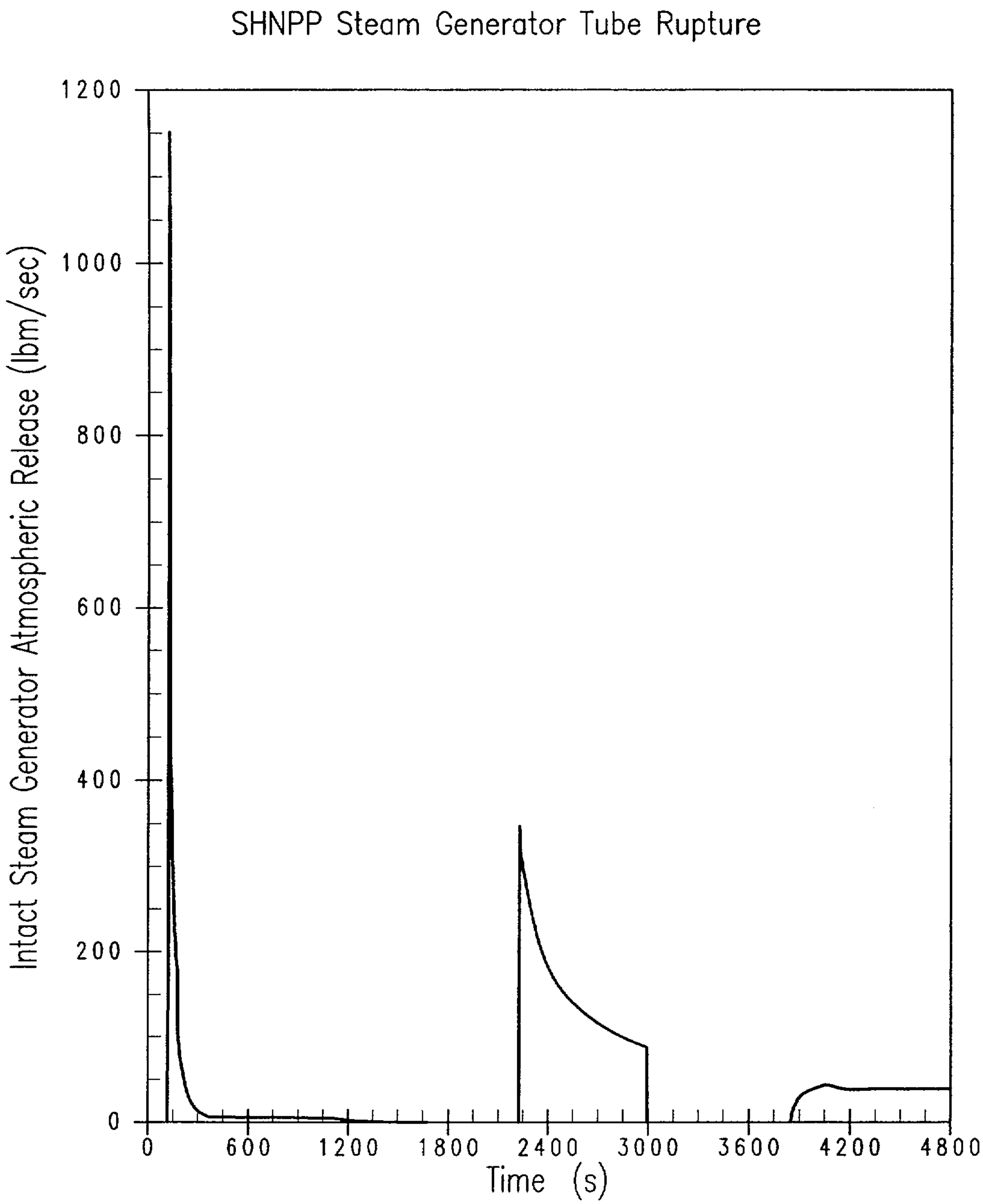

Figure 6.3.2-10
Intact SGs Mass Release Rate to the Atmosphere –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

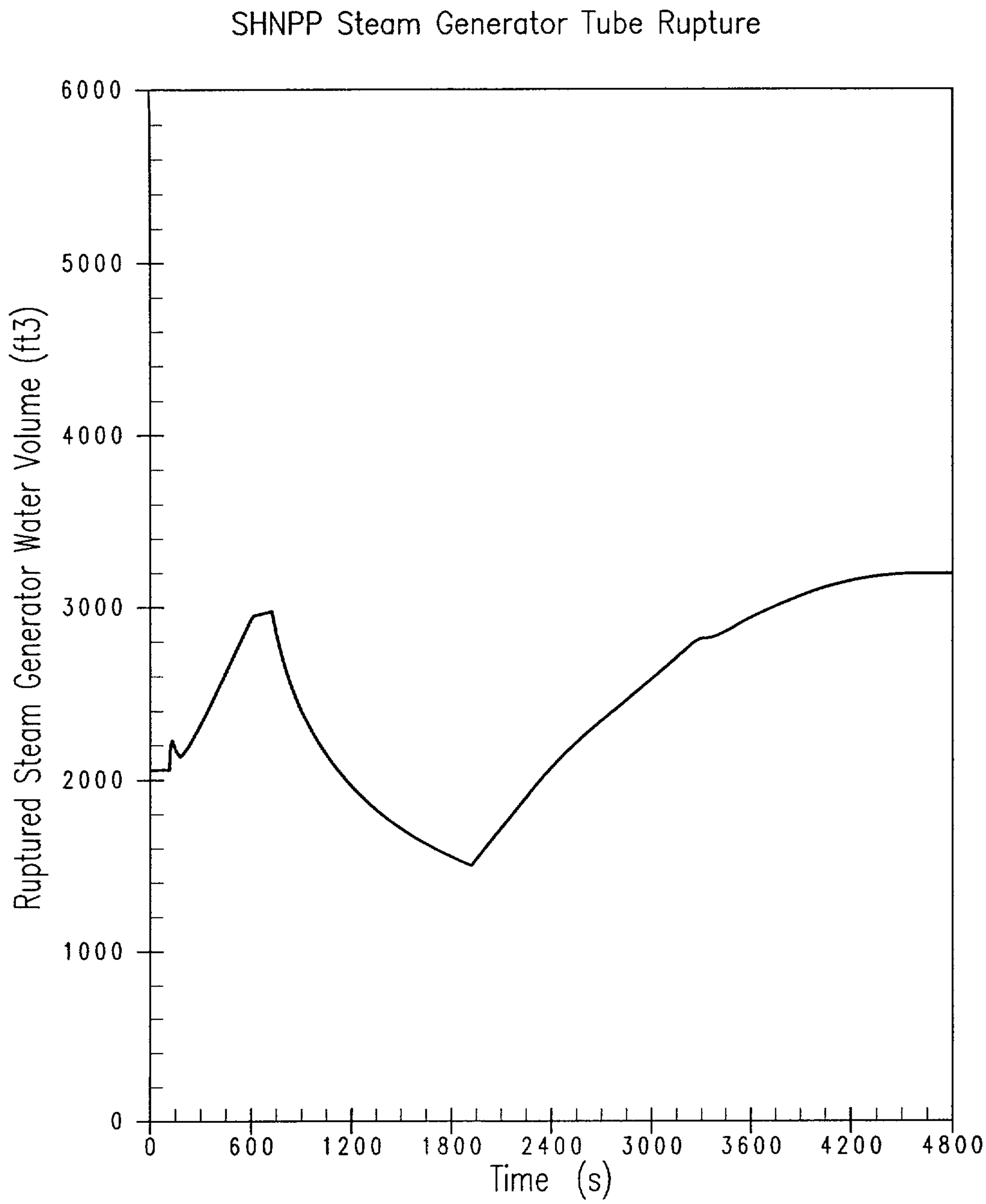

Figure 6.3.2-11
Ruptured SG Water Volume –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

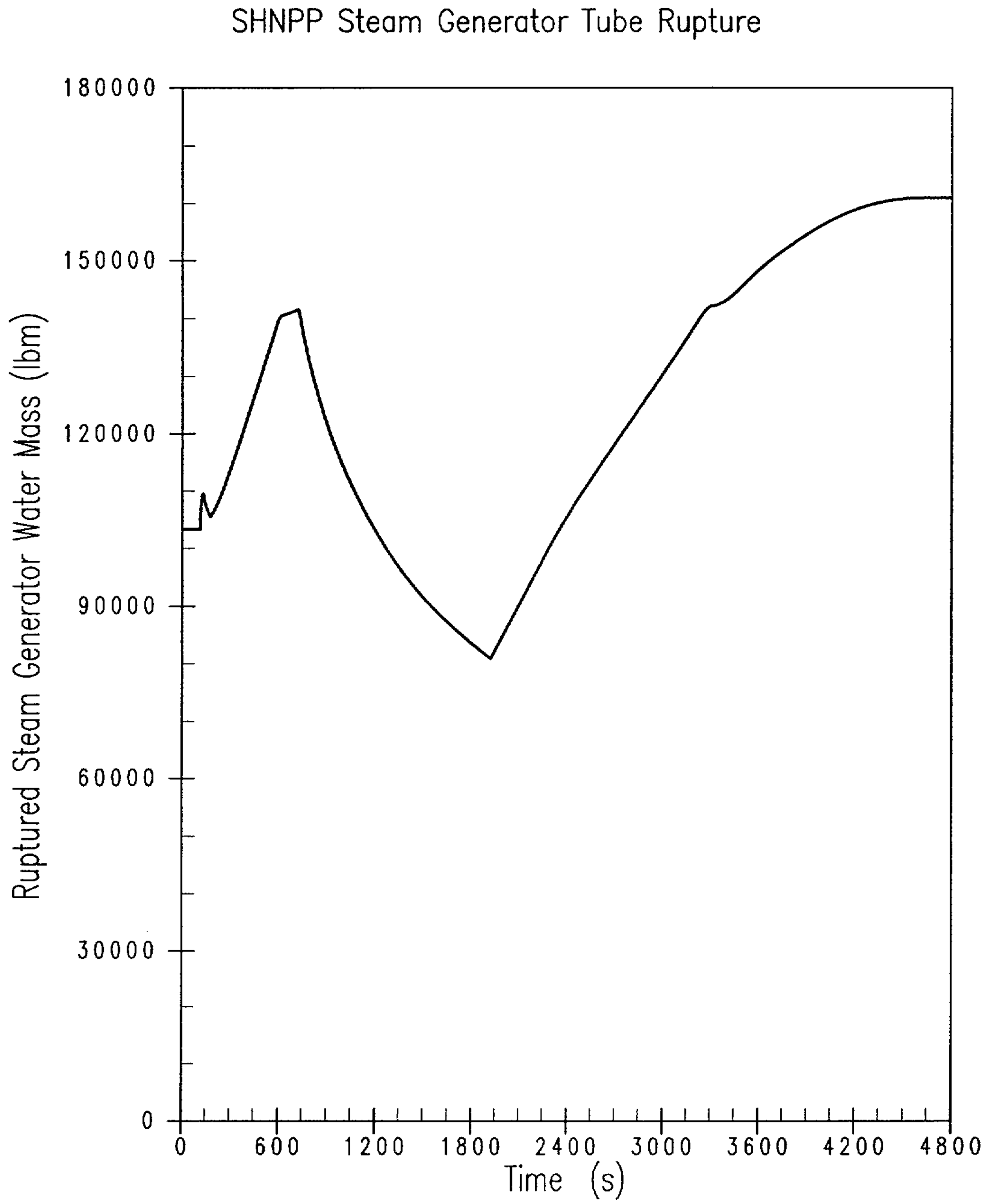

Figure 6.3.2-12
Ruptured SG Water Mass –
Offsite Radiation Dose Analysis (NSSS Power of 2912.4 MWt)

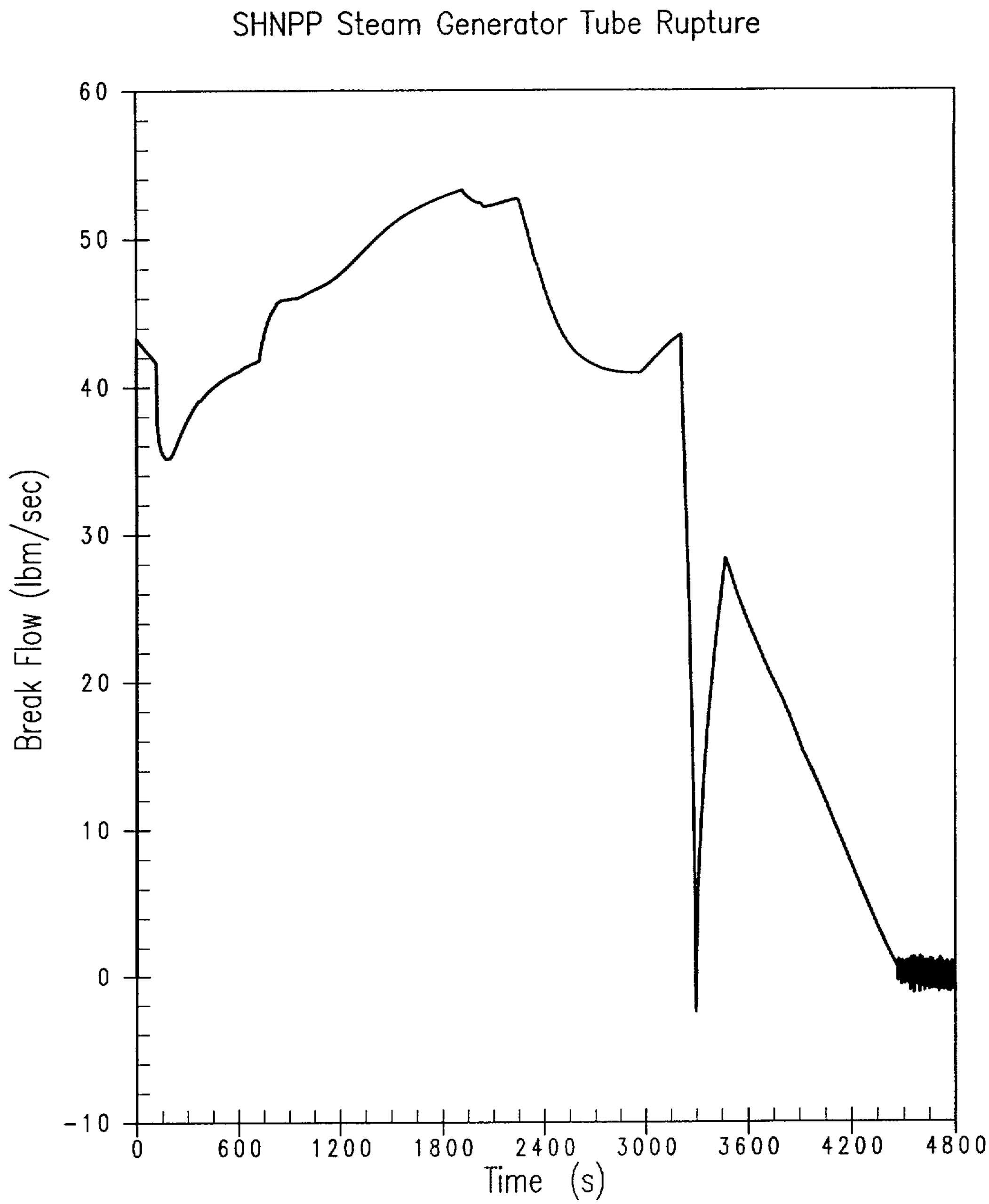

Figure 6.3.2-13
Primary to Secondary Break Flow –
Offsite Radiation Dose Analysis (NSSS power of 2787.4 MWt)

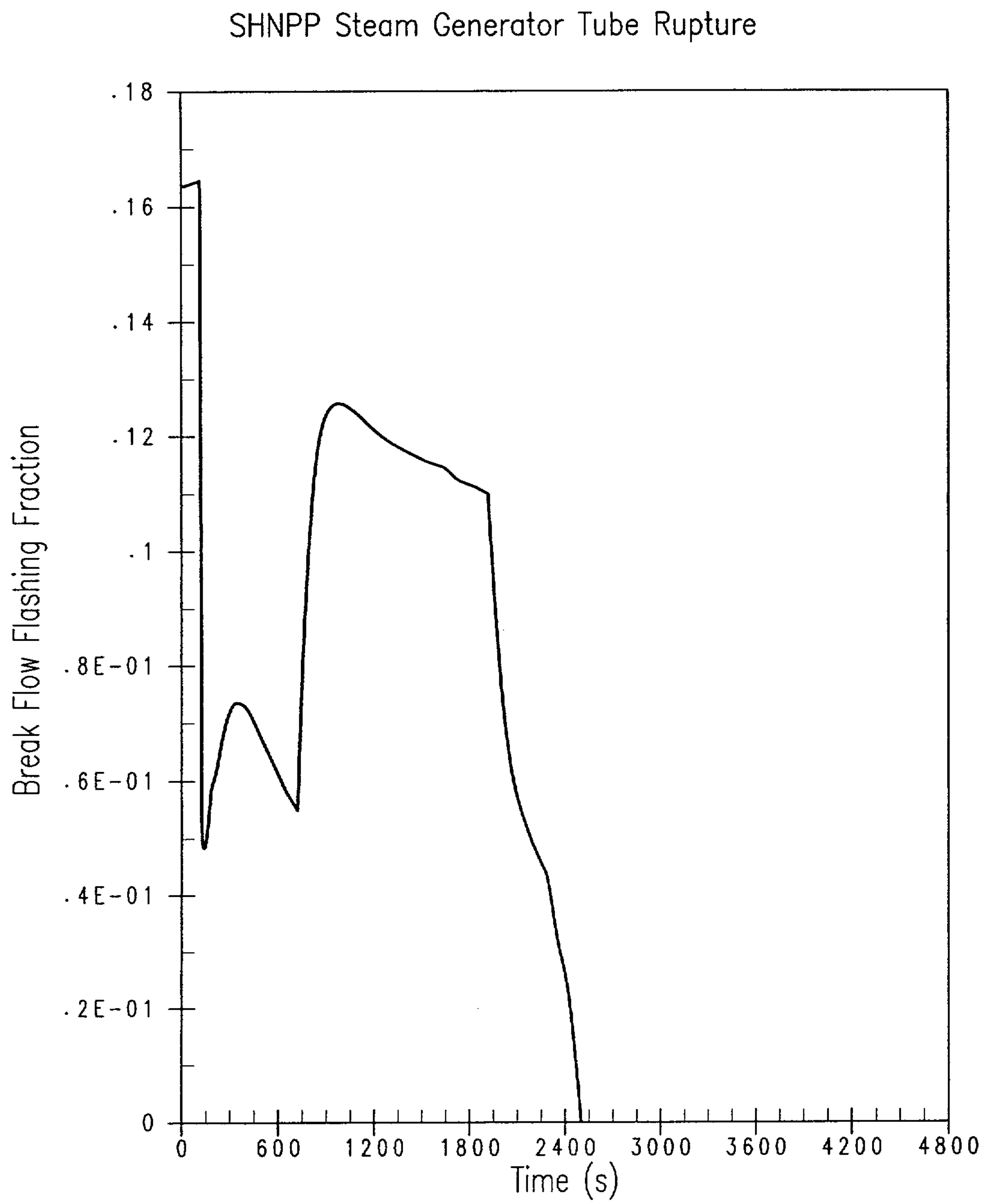

Figure 6.3.2-14
Break Flow Flashing Fraction –
Offsite Radiation Dose Analysis (NSSS Power of 2787.4 MWt)

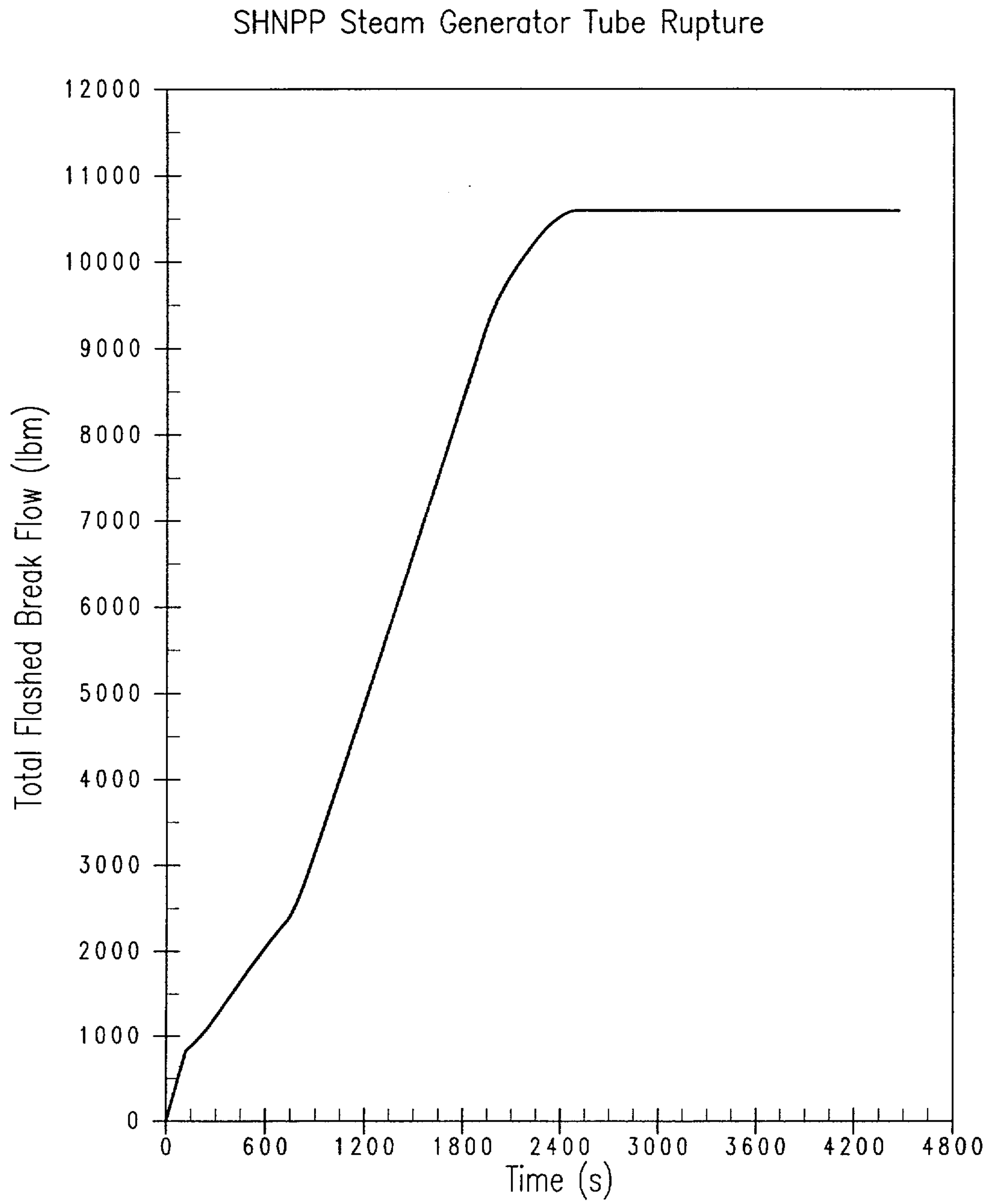

Figure 6.3.2-15
Total Flashed Break Flow –
Offsite Radiation Dose Analysis (NSSS Power of 2787.4 MWt)

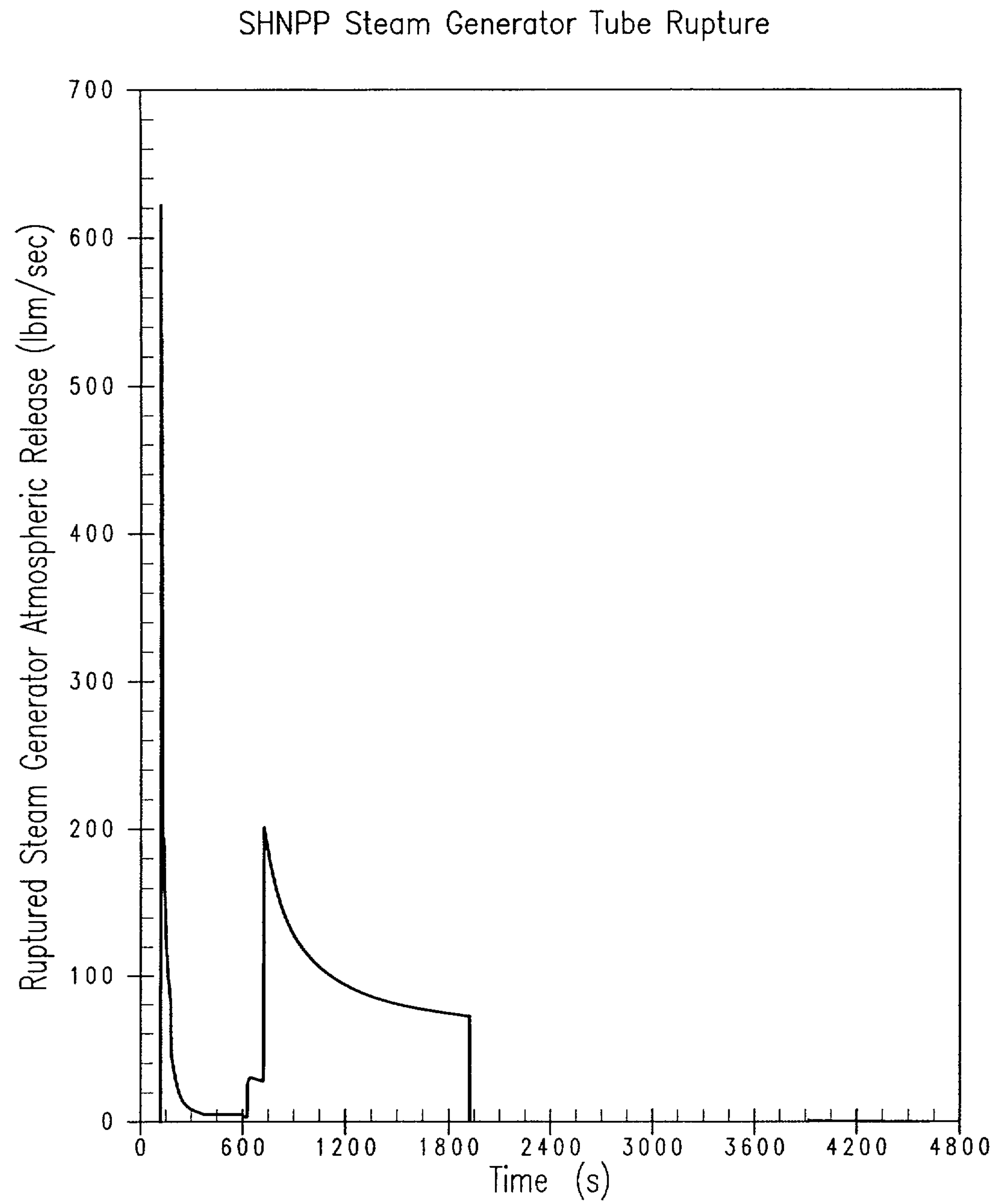

Figure 6.3.2-16
Ruptured SG Mass Release Rate to the Atmosphere –
Offsite Radiation Dose Analysis (NSSS Power of 2787.4 MWt)

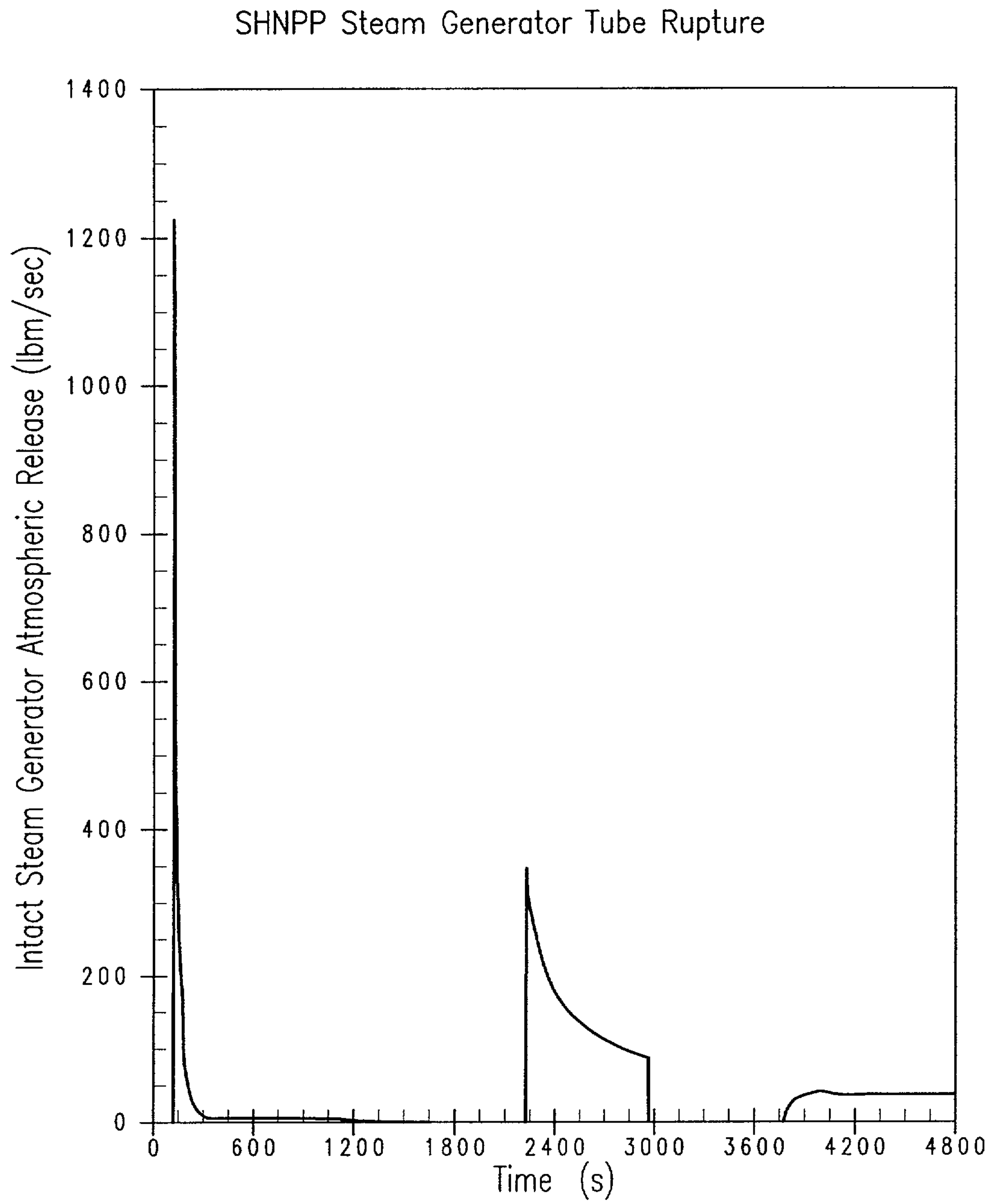

Figure 6.3.2-17
Intact SGs Mass Release Rate to the Atmosphere –
Offsite Radiation Dose Analysis (NSSS Power of 2787.4 MWt)

6.3.2 Radiological Consequences Analysis

6.3.2.1 Introduction

The evaluation of the radiological consequences of a steam generator tube rupture (SGTR) assumes that the reactor has been operating at the Technical Specification limits for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radio-nuclides in the reactor coolant and in the secondary coolant. Radio-nuclides from the primary coolant enter the steam generator, via the ruptured tube and primary to secondary leakage, and are released to the atmosphere through the steam generator safety or power operated relief valves (PORVs) and via the condenser air ejector exhaust.

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

The most recent SGTR radiological consequences analysis performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 1 and 2) were performed using the analysis methodology developed in Supplement 1 to WCAP-10698 (Reference 3). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in a Safety Evaluation Report (SER) dated December 17, 1985. The SGTR radiological consequences analysis was performed in support of the HNP Model Delta 75 replacement steam generator program using this methodology with some variations. These variations in methodology reflect the latest accepted methods and are identified in this report.

Section 6.3.2 of this report presents the mass releases for the SGTR event assuming failure and isolation of the ruptured steam generator PORV for analyses modeling the Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt and at the current NSSS power of 2787.4 MWt. The resulting offsite and control room doses are calculated in this section.

This section includes the methods and assumptions used to analyze the radiological consequences of the SGTR event, as well as the calculated results.

6.3.2.2 Description of Analyses and Evaluations

Major assumptions and parameters are summarized in Table 6.3.3-1.

6.3.3.2.1 Source Term Assumptions

The radio-nuclide concentrations in the primary and secondary system, prior to and following the SGTR, are determined as follows.

1. The iodine concentrations in the reactor coolant are based upon pre-accident and accident initiated iodine spikes as outlined in the Standard Review Plan (SRP) Section 15.6.3 (Reference 4).

   a. Pre-accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration to 60 μCi/gm of Dose Equivalent (D.E.) I-131.

   b. Accident-Initiated Spike - The primary coolant iodine concentration is initially at the Technical Specification limit, specified in μCi/gm of D.E. I-131. Following the primary system depressurization and reactor trip associated with the SGTR, an iodine spike is initiated in the primary system. This spike increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. This release rate (the equilibrium iodine appearance rate) is calculated to match the rate of iodine removal from the RCS. Iodine removal from the RCS is the combination of decay, leakage and cleanup.

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

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

4. The initial concentration of noble gases in the reactor coolant is based on one-percent defective fuel, which corresponds to the Technical Specification limit of 100/E-bar.

5. No noble gases are present in the secondary system at the start of the event.

The concentration of iodine and noble gas nuclides in the reactor coolant system (RCS) has been calculated based on a one-percent fuel defect level for the SGR/Uprating program. The concentration data presented in Table 6.3.3-2 is used in the SGTR analysis (taken from Section 7.6 of this report). Although this source term data was calculated at the uprated power, it is used in all SGTR dose calculations presented in this report. Since the iodine activity is defined in D.E. I-131, and the distribution of iodine isotopes is not sensitive to small changes in core power, using iodine concentrations based on the uprated power has no impact on the results. The noble gas activity is approximately proportional to the power level assumed in the calculations. Use of the uprated power noble gas activities is conservative and since the noble gas doses are not limiting for a SGTR, they have only a small impact on the results.

The conversion from the one-percent fuel defect values in Table 6.3.3-2 to DE I-131 employs dose conversion factors (DCFs). DCFs are also used in calculating the dose resulting from iodine releases. In the previous analyses the thyroid dose conversion factors are from Regulatory Guide (RG) 1.109 (Reference 5). In order to be consistent with current analysis techniques, thyroid dose conversion factors from International Commission on Radiological Protection (ICRP)-30 (Reference 6) is used in this analysis. These DCFs are used in the calculation of the initial RCS iodine concentrations. The ICRP-30 thyroid dose conversion factors used in the analysis are presented in Table 6.3.3-3.

The current Technical Specification limit for RCS iodine activity is 1.0 μCi/gm D.E. I-131. The HNP Technical Specification limit for RCS iodine activity is being reduced to 0.35 μCi/gm D.E. I-131 for this submittal.

The spike model used in the previous analyses calculated equilibrium iodine appearance rates based on a letdown flow of 60 gpm with 90 percent cleanup. The spike appearance rate is 500 times the equilibrium appearance rate. The conservative spike model calculates the equilibrium iodine appearance rates based on a letdown flow of 120 gpm with perfect cleanup. This flow is conservatively increased by 10 percent to cover uncertainties in the flow. In addition, a total of 42-gpm leakage from the RCS allowed by the Technical Specifications (which also remove iodine from the RCS) is considered in the calculations. The effective letdown flow increases from 54 gpm with the previous non-conservative spike model to 174 gpm with the conservative spike model. The 174 gpm is the total of 120 gpm letdown flow with perfect cleanup increased by 10 percent to 132 gpm (to cover uncertainty), 10 gpm identified leakage from the RCS, 1 gpm unidentified leakage from the RCS, and 31 gpm controlled leakage.

In all cases the spike is allowed to continue until 5 hours from the start of the event. This bounds the time calculated for all iodine initially contained in the gap of the defective fuel to be transferred to the coolant at the spike appearance rate being modeled. In the previous analyses, the spike was assumed to be terminated at 2.78 hours. The spike duration was extended in response to NRC comments on recent analyses performed for other plants. This has little impact on the SGTR analysis, since the majority of the iodine releases end shortly after the ruptured steam generator PORV is isolated at about 30 minutes from the start of the event.

The initial RCS iodine activities used in the analyses are presented in Table 6.3.3-4. The iodine appearance rates used in the analyses are presented in Table 6.3.3-5.

6.3.3.2.2 Dose Calculation Assumptions

Offsite power is assumed to be lost at reactor trip. This assumption was used in the thermal-hydraulic analysis (Section 6.3.2) to maximize break flow and steam release though the ruptured steam generator PORV. Prior to reactor trip, a condenser iodine partition factor of 0.01 is assumed. After reactor trip and loss of offsite power, flow to the condenser is isolated. This condenser iodine partition factor is consistent with the NUREG-0800 SRP 15.6.3 (Reference 4) steam/water partition coefficient.

The iodine transport model used in this analysis accounts for break flow flashing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. The fraction of primary coolant iodine that is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate. The 0.01 steam/water partition coefficient from NUREG-0800 SRP 15.6.3 (Reference 4) is used. Droplet removal by the dryers is conservatively neglected.

In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface was not calculated and was conservatively neglected. Although this removal was calculated and credited in the previous analyses using a model based on that proposed in NUREG-0409 (Reference 7), it is no longer considered in standard Westinghouse analyses.

Average disintegration energies for nuclides are found in Table 6.3.3-6. These are used in calculating the gamma doses and control room beta skin doses.

All of the iodine in the flashed break flow is assumed to be transferred instantly out of the steam generator to the atmosphere.

The issue of tube bundle uncovery was considered in a Westinghouse Owners Group (WOG) program (Reference 8). The WOG program concluded that the effect of tube uncovery is essentially negligible for the limiting SGTR transient. The WOG program concluded that the steam generator tube uncovery issue could be closed without any further investigation or generic restrictions. The NRC review of the WOG submittal (Reference 9) concluded "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue resolved." This modeling is different from that used in the previous analyses. Those analyses were completed prior to the resolution of the tube uncovery issue and conservatively modeled the direct release of all iodine transferred to the ruptured steam generator in the break flow when the tubes were assumed to be uncovered.

Since there is no penalty taken for tube uncovery and no iodine scrubbing is credited, the location of the tube rupture is not significant for the radiological analysis. The thermal and hydraulic analysis presented in Section 6.3.2 has conservatively addressed the issue of the location of the tube rupture in the calculations of break flow and flashing of break flow.

No credit is taken for the radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the control room, site boundary, or outer boundary of the low population zone (LPZ).

All noble gases in the break flow and primary-to-secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere.

Iodine and noble gas decay constants are presented in Table 6.3.3-7. These decay constants were calculated from half-lives given in Reference 10.

Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table 6.3.3-8. The offsite breathing rates were obtained from NRC RG 1.4 (Reference 11) and the control room breathing rates and occupancy factors are from Murphy-Campe (Reference 12).

Offsite Dose Calculation Model

Thyroid and whole body gamma doses are calculated for 2 hours at the site boundary. At the LPZ thyroid and whole body gamma doses are calculated up to the time all releases are terminated, which is the RHR cut in time used in the thermal and hydraulic analysis. Offsite thyroid doses are calculated using the following equation.

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

where:

D_{Th} = thyroid dose via inhalation (rem)

DCF_i = thyroid dose conversion factor via inhalation for isotope i (rem/Ci) (Table 6.3.3-3)

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

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 6.3.3-8)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 6.3.3-8)

Offsite whole body gamma doses are calculated using the following equation:

$$D_{WB} = 0.25 \sum_i \left[E_{\gamma i} \left(\sum_j (IAR)_{ij} \; (\chi / Q)_j \right) \right]$$

where:

D_{WB} = whole body dose via cloud immersion (rem)

$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis) (Table 6.3.3-6)

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

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 6.3.3-8)

The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases. This is more limiting than the previous calculations which only considered the contribution from noble gases. In order to allow a comparison to the previously calculated results, the whole body doses resulting from the noble gases alone are also calculated.

Control Room Dose Calculation Models

Thyroid, whole body gamma, and beta skin doses are calculated for 30 days in the control room. Although all releases are terminated when the RHR system is put in service, the calculation is continued to account for additional doses due to continued occupancy.

The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room and the control room filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analysis are presented in Table 6.3.3-9. Control room thyroid doses are calculated using the following equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j \right) \right]$$

where:

D_{Th} = thyroid dose via inhalation (rem)

DCF_i = thyroid dose conversion factor via inhalation for isotope i (rem/Ci) (Table 6.3.3-3)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 6.3.3-8)

Control room whole body doses are calculated using the following equation:

$$D_{WB} = 0.25 * \left(\frac{1}{GF} \right) * \sum_i E_{\gamma i} \left(\sum_j Conc_{ij} \right)$$

where:

D_{WB} = whole body dose via cloud immersion in rem

GF = geometry factor, calculated based on Reference 12, using the equation

$$GF = \frac{1173}{V^{0.338}}$$ where V is the control room volume in ft^3

$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis) (Table 6.3.3-6)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

Control room skin doses are calculated using the following equation:

$$D_\beta = 0.23 * \sum_i E_{\beta i}\left(\sum_j Conc_{ij}\right)$$

where:

D_β = whole body dose via cloud immersion (rem)

$E_{\beta i}$ = average beta disintegration energy for isotope i (Mev/dis) (Table 6.3.3-6)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

6.3.3.2.3 Mass Transfer Assumptions

Break flow, flashing break flow and steam releases from the intact and ruptured steam generators are modeled using data from the thermal and hydraulic analysis in Section 6.3.2 of this report.

A total primary to secondary leak rate is assumed to be 1.0 gpm. The leak is assumed to be distributed with 0.7 gpm to the two intact steam generators and 0.3 gpm to the ruptured steam generator. The leakage to the intact steam generators is assumed to persist for the duration of the accident. This modeling is consistent with the previous analyses. Atmospheric conditions are assumed in determining the density for this leakage.

In addition to the releases calculated in the thermal hydraulic analysis presented in Section 6.3.2, steam released from the ruptured steam generator to the turbine driven auxiliary feedwater (TDAFW) pump is considered in the dose analysis. A flow of 41,310 lbm/hr is considered from the time of auxiliary feedwater initiation until the ruptured steam generator is isolated. The iodine contained in this steam, determined from the steam generator activity and the water/steam partition coefficient of 100, is assumed to be released directly to the atmosphere. This flow was not modeled in the previous analyses and is not required by the Reference 3 methodology. Inclusion of this flow is conservative since it is assumed to be released directly to the atmosphere.

6.3.3.3 Description of Analyses Cases

Offsite and control room doses are calculated for the two thermal hydraulic analyses presented in Section 6.3.2 of this report. One set of dose calculations corresponds to the analysis performed at the uprated NSSS power of 2912.4 MWt with Model Delta 75 replacement steam generators. For this set the mass transfer data is taken from Table 6.3.2-2 and Figures 6.3.2-6, 6.3.2-8, 6.3.2-9, and 6.3.2-10. A second set of dose calculations corresponds to the analysis performed at the current NSSS power of 2787.4 MWt with Model Delta 75 replacement steam generators. For this set the mass transfer data is taken from Table 6.3.2-3 and Figures 6.3.2-13, 6.3.2-15, 6.3.2-16, and 6.3.2-17.

Each set of calculations determines the thyroid doses based on a pre-accident iodine spike of 60.0 μCi/gm D.E. I-131 primary coolant activity. Thyroid doses for two assumed accident-initiated iodine spike appearance rates are calculated for each set of thermal hydraulic results, as described in Section 6.3.3.2.1. Both spike assumptions consider 0.1 μCi/gm D.E. I-131 secondary activity.

The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases, as described in Section 6.3.3.2.2. In the Reference 1 and 2 analyses the whole body doses reported were based solely on the noble gas contribution consistent with the Reference 3 methodology. The current industry practice is to include the iodine contribution in the whole body doses. The whole body doses are calculated with the limiting iodine releases (either pre-accident spike or accident-initiated iodine spike).

6.3.3.4 Acceptance Criteria

The acceptance limits for doses must be satisfied for a SGTP. The offsite dose limits are specified in NUREG-0800 SRP 15.6.3 (Reference 4). The doses at the site boundary (SB) and the LPZ for a SGTR with an assumed pre-accident iodine spike must be within the 10 CFR 100 limits, (i.e., less than 300 rem thyroid and 25 rem whole body). The doses at the SB and the LPZ for a SGTR with an assumed accident-initiated iodine spike must be within a small fraction (10 percent) of the 10 CFR 100 limits, i.e., less than 30-rem thyroid and 2.5-rem whole body. The control room dose limits are specified in NUREG-0800 SRP 6.4 (Reference 13) based on General Design Criteria (GDC) 19. Doses in the control room must be less than 30-rem thyroid, 5-rem whole body, and 30-rem beta-skin.

The site boundary doses are calculated for 2 hours. The LPZ doses are calculated up to the time all releases are terminated, which is the Residual Heat Removal (RHR) cut in time (8 hours) used in the thermal and hydraulic analysis in Section 6.3.2. The control room doses are calculated for 30 days.

6.3.3.5 Results

The pre-accident iodine spike thyroid doses for the SGTR analysis with Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt and at the current NSSS power level of 2787.4 MWt are tabulated in Table 6.3.3-10. The table includes the most recent Westinghouse reported doses (from Reference 2) and the applicable limit. The results in the table demonstrate that the SGR/Uprating does not result in an increase in the previous pre-accident iodine spike thyroid doses. The applicable limits are met.

Table 6.3.3-11 presents the accident-initiated iodine spike doses calculated based on a primary coolant iodine limit of 0.35 μCi/gm D.E. I-131, and spike appearance rates calculated with conservative assumptions. The results in the table demonstrate that the applicable limits are met. The reduction in allowable primary coolant iodine activity to 0.35 is sufficient to offset the penalty associated with the revised spike appearance rate calculations.

Table 6.3.3-12 presents the whole body doses calculated using only noble gas releases, consistent with those reported in Reference 2. The table includes the most recent Westinghouse reported doses (from Reference 2) and the applicable limit. The results in the table demonstrate that the SGR/Uprating does not result in an increase in the previous whole body doses. These results provide a direct comparison to the current analysis of record. Table 6.3.3-13 presents the whole body doses calculated including the contribution from iodines. The iodine contribution from the limiting iodine spike case, determined to be the pre-accident spike, is used. The results in the table demonstrate that the applicable limits are met.

Table 6.3.3-14 presents the control room beta skin doses, including the allowable guideline value. The iodine contribution from the limiting iodine spike case is used. The results in the table demonstrate that the applicable limits are met.

6.3.3.6 Conclusions

The potential radiological consequences of a SGTR were evaluated for HNP in support of the SGR/Uprating program. Since it was analyzed in Section 6.3.1 that steam generator overfill will not occur for a design basis SGTR, an analysis was performed to determine the offsite radiation doses assuming the limiting single failure for offsite doses. The thermal hydraulic results from this analysis are presented in Section 6.3.2. The resulting doses at the exclusion area boundary, low population zone, and control room (presented in Section 6.3.3) are within the allowable guidelines. The analysis also demonstrated that the doses do not increase, relative to previously reported values (Reference 2), as a result of the SGR/Uprating.

6.3.3.7 References

1. WCAP-12403, "LOFTTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.
2. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz, November 1992.
3. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986.
4. NUREG-0800, Standard Review Plan Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," Rev. 2, July 1981.
5. Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," US Nuclear Regulatory Commission, October 1977.
6. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979.
7. NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture," A. K. Postma, P. S. Tam.
8. WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992.
9. Letter from Robert C. Jones to Lawrence A. Walsh, "Westinghouse Owners Group-Steam Generator Tube Uncovery Issue," March 10, 1993.
10. ENDF-223, "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," T. R. England and R. E. Schenter, October 1975.
11. NRC Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors," June 1974.
12. K. G. Murphy and K. W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," published in Proceedings of 13th AEC Air Cleaning Conference, Atomic Energy Commission (now NRC), August 1974.
13. NUREG-0800, Standard Review Plan 6.4, "Control Room Habitability System," Revision 2, July 1981.

Table 6.3.3-1
Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture

I. Source Data	
A. Core Power Level, MWt	2900
B. Reactor Coolant Iodine Activity:	
1. Accident-Initiated Spike	The initial RC iodine activities are presented in Table 6.3.3-4. The iodine appearance rates assumed for the accident-initiated spike are presented in Table 6.3.3-5.
2. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gm of D.E. I-131 are presented in Table 6.3.3-4.
C. Noble Gas Activity	Primary coolant noble gas activities based on 1-percent fuel defects are presented in Table 6.3.3-2. No noble gases are contained in the secondary system.
D. Secondary System Initial Activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table 6.3.3-4.
E. Reactor Coolant Initial Mass, grams	
Uprated Power	1.73×10^{8}
Current Power	1.73×10^{8}
F. Steam Generator Initial Mass (each), grams	
Uprated Power	4.34×10^{7}
Current Power	4.38×10^{7}
G. Offsite power	Lost at time of reactor trip
H. Primary-to-Secondary Leakage Duration for Intact SG, hours	8
I. Species of Iodine	100 percent elemental

Table 6.3.3-1 (Cont.) Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture	
II. Activity Release Data	
A. Ruptured Steam Generator	
1. Rupture Flow	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-6
Current Power	See Table 6.3.2-3 & Figure 6.3.2-13
2. Flashed Rupture Flow	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-8
Current Power	See Table 6.3.2-3 & Figure 6.3.2-15
3. Steam Releases	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-9
Current Power	See Table 6.3.2-3 & Figure 6.3.2-16
	In both cases an additional 41,310 lbm/hr to TDAFW pump is modeled until ruptured SG isolation.
4. Iodine Partition Factor for Rupture Flow	
Non-flashed	100
Flashed	1.0
B. Intact Steam Generators	
1. Primary-to-Secondary Leakage, gpm	0.7
2. Steam Releases	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-10
Current Power	See Table 6.3.2-3 & Figure 6.3.2-17
3. Iodine Partition Factor	100
C. Condenser	
1. Iodine Partition Factor	100
D. Atmospheric Dispersion Factors	See Table 6.3.3-8

Table 6.3.3-2
Reactor Coolant Fission Product Specific Activity Based on 1-Percent Fuel Defects

Nuclide	Specific Activity (μCi/gm)
I-131	1.71E+00
I-132	2.47E+00
I-133	7.23E+00
I-134	5.67E-01
I-135	1.84E+00
Kr-85m	1.7
Kr-85	10.6
Kr-87	1.1
Kr-88	3.2
Xe-131m	3.4
Xe-133m	4.9
Xe-133	276.4
Xe-135m	0.4
Xe-135	8.5
Xe-138	0.6

Table 6.3.3-3 **Thyroid Dose Conversion Factors (Reference 6*)**	
Nuclide	**DCF (Rem/Curie)**
I-131	1.07 x 10^{6}
I-132	6.29 x 10^{3}
I-133	1.81 x 10^{5}
I-134	1.07 x 10^{3}
I-135	3.14 x 10^{4}

*Reference 6 provides the dose conversion factors in units of sievert/becquerel.

Table 6.3.3-4
Iodine Specific Activities (μCi/gm) in the Primary Coolant Based on 0.35 and 60.0 μCi/gm of D.E. I-131 and in the Secondary Coolant Based on 0.1 μCi/gm of D.E. I-131

	Primary Coolant		**Secondary Coolant**
Nuclide	**0.35 μCi/gm**	**60 μCi/gm**	**0.1 μCi/gm**
1-131	0.200	34.20	0.0570
1-132	0.288	49.38	0.0823
1-133	0.843	144.48	0.2408
1-134	0.066	11.34	0.0189
1-135	0.215	36.78	0.0613

Table 6.3.3-5 Iodine Spike Appearance Rates (Curies/Minute) Based on 0.35 μCi/gm of D.E. I-131 Primary Coolant Activity						
Primary Activity	**Assumptions**	**I-131**	**I-132**	**I-133**	**I-134**	**I-135**
0.35 μCi/gm D.E. I-131	Conservative	67.0	220.5	318.0	97.0	103.5

Table 6.3.3-6 Disintegration Energies (Reference 10)		
Nuclide	**Gamma Disintegration Energy (Mev/Dis)**	**Beta Disintegration Energy (Mev/Dis)**
I-131	0.38	0.19
I-132	2.2	0.52
I-133	0.6	0.42
I-134	2.6	0.69
I-135	1.4	0.43
Kr-85m	0.16	0.25
Kr-85	0.0023	0.25
Kr-87	0.79	1.3
Kr-88	2.2	0.25
Xe-131m	0.0029	0.16
Xe-133m	0.02	0.21
Xe-133	0.03	0.15
Xe-135m	0.43	0.099
Xe-135	0.25	0.32
Xe-138	1.2	0.66

Table 6.3.3-7
Decay Constants (Reference 10)

Nuclide	Decay Constant (1/hr)
I-131	0.00359
I-132	0.303
I-133	0.0333
I-134	0.791
I-135	0.105
Kr-85m	0.155
Kr-85	7.37E-6
Kr-87	0.547
Kr-88	0.248
Xe-131m	0.00241
Xe-133m	0.0130
Xe-133	0.00546
Xe-135m	2.72
Xe-135	0.0756
Xe-138	2.93

Table 6.3.3-8 Atmospheric Dispersion Factors and Breathing Rates						
Time (hours)	**Exclusion Area Boundary χ/Q (sec/m^3)**	**Low Population Zone χ/Q (sec/m^3)**	**Control Room χ/Q (sec/m^3)**	**Offsite Breathing Rate (m^3/sec)**	**Control Room Breathing Rate (m^3/sec)**	**Control Room Occupancy Factor**
0 - 2	6.17×10^{-4}	1.4×10^{-4}	4.08×10^{-3}	3.47×10^{-4}	3.47×10^{-4}	1.0
2 - 8	----	1.4×10^{-4}	4.08×10^{-3}	3.47×10^{-4}	3.47×10^{-4}	1.0
8 - 24	----	----	1.16×10^{-3}	1.75×10^{-4}	3.47×10^{-4}	1.0
24 - 96	----	----	3.25×10^{-4}	2.32×10^{-4}	3.47×10^{-4}	0.6
> 96	----	----	1.23×10^{-5}	2.32×10^{-4}	3.47×10^{-4}	0.4

Table 6.3.3-9 **Control Room Model**	
Control Room Isolation Signal Generated	Time of SI signal from Section 6.3.2
Delay in Control Room Isolation After Isolation Signal is Generated	30 Seconds
Control Room Volume	71000 ft^3
Control Room HVAC Ductwork Volume	2704 ft^3
Control Room Unfiltered In-Leakage	80 cfm
Control Room Unfiltered Inflow	
Normal Mode	1050 cfm
Emergency Mode	0 cfm
Control Room Filtered Inflow	
Normal Mode	0 cfm
Emergency Mode	400 cfm
Control Room Filtered Recirculation	
Normal Mode	0 cfm
Emergency Mode	3600 cfm
Control Room Filter Efficiency	99%

Table 6.3.3-10 Pre-Accident Iodine Spike Thyroid Doses				
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Reference 2 Doses (Rem)**	**Allowable Guideline Value**
Pre-Accident Iodine Spike - Thyroid				
Exclusion Area Boundary (0-2 hr.)	57.74	56.47	90.4	300
Low Population Zone (0-8 hr.)	13.84	13.51	20.7	300
Control Room (0-30 Days)	10.93	10.43	Not Calculated	30

Table 6.3.3-11 Accident-Initiated Iodine Spike Thyroid Doses Iodine Spike Appearance Rates Based on 0.35 μCi/gm of D.E. I-131 Primary Coolant Activity			
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Allowable Guideline Value**
Accident-Initiated Iodine Spike - Thyroid			
Exclusion Area Boundary (0-2 hr.)	15.86	15.58	30
Low Population Zone (0-8 hr.)	4.07	3.97	30
Control Room (0-30 Days)	2.54	2.48	30

Table 6.3.3-12 Whole Body Doses From Noble Gases				
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Reference 2 Doses (Rem)**	**Allowable Guideline Value**
Whole Body Gamma Dose				
Exclusion Area Boundary (0-2 hr.)	0.170	0.169	0.2	2.5
Low Population Zone (0-8 hr.)	0.039	0.038	0.1	2.5
Control Room (0-30 Days)	0.036	0.036	Not Calculated	5

Table 6.3.3-13 Total Whole Body Doses			
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Allowable Guideline Value**
Whole Body Gamma Dose			
Exclusion Area Boundary (0-2 hr.)	0.35	0.34	2.5
Low Population Zone (0-8 hr.)	0.08	0.08	2.5
Control Room (0-30 Days)	0.04	0.04	5

Table 6.3.3-14 Control Room Skin Doses				
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Reference 2 Doses (Rem)**	**Allowable Guideline Value**
Beta Skin Dose				
Control Room (0-30 Days)	3.04	3.01	Not Calculated	30

6.4 LOCA Mass and Energy Releases

The uncontrolled release of pressurized high temperature reactor coolant, or loss-of-coolant accident (LOCA), results in release of steam and water into the containment. This, in turn, results in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. There are both long and short-term issues relative to a postulated LOCA that must be considered for the SGR/Uprating for the Harris Nuclear Plant (HNP).

The long-term LOCA mass and energy (M&E) releases, addressed in Section 6.4.1, are utilized as input to the containment integrity analysis, which demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA.

The short-term LOCA-related M&E releases, addressed in Section 6.4.2, are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high energy line pipe rupture within that subcompartment.

6.4.1 Long-Term LOCA M&E Releases

6.4.1.1 Introduction

The limiting long-term LOCA mass and energy releases are analyzed to approximately 3×10^7 seconds, or one year, and are utilized as input to the containment integrity analysis. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and limiting the temperature excursion to less than the Environmental Qualification (EQ) acceptance limits. The containment safeguards systems must also be capable of limiting the peak containment pressure to less than the Integrated Leak Rate Test (ILRT) pressure and reducing the pressure to less than 50 percent of the calculated pressure in 24 hours. For the SGR/Uprating program, Westinghouse generated the M&E releases using the March 1979 model, described in Reference 1, which includes the NRC review and approval letter. This methodology has previously been applied to the HNP (Reference 2). The long-term LOCA M&E releases generated by Westinghouse for this program have been provided to CP&L for use in the containment integrity analysis and EQ reviews. (See the BOP Licensing Report.)

Section 6.4.1 addresses the long-term LOCA M&E releases for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot-leg (DEHL) rupture break cases.

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the Reactor Coolant System (RCS) operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of (+6.0°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+51 psi).

All input parameters are chosen consistent with accepted analysis methodology. Some of the most-critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed next. Tables 6.4.1-1, 6.4.1-2 and 6.4.1-3 present key data assumed in the analysis.

The core rated power of 2900 MWt adjusted for calorimetric error (+2 percent of power) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The selection of 2250 psia as the limiting pressure is considered to affect 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.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term M&E release calculations.

The selection of the fuel design features for the long-term M&E release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The margin in core-stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3 percent (1.6 percent allowance for thermal expansion and 1.4 percent for uncertainty) is modeled.

A uniform steam generator tube plugging (SGTP) level of 0 percent is modeled. This assumption maximizes the reactor coolant volume and fluid release by considering the RCS fluid in all SG tubes. During the post-blowdown period the steam generators are active heat sources, as significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0-percent SGTP assumption maximizes heat transfer area and therefore, the transfer of secondary heat across the SG tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the pressure drop upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the level of SGTP.

Regarding safety injection flow, the M&E release calculation considered configurations/failures to conservatively bound respective alignments. These cases include (1) a Minimum Safeguards case (one Charging/Safety Injection pump [CH/SI] and one Low Head Safety Injection [LHSI] pump) and (2) a Maximum Safeguards case (two CH/SI and two LHSI pumps).

The following assumptions were employed to ensure that the M&E releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS (100-percent full-power conditions)

2. Allowance for RCS temperature uncertainty (+6.0°F)

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

4. Core rated power of 2900 MWt

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

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

7. Allowance in core-stored energy for effect of fuel densification

8. A margin in core-stored energy (+15 percent to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+51 psi)

10. A maximum containment backpressure equal to design pressure (45 psig)

11. Allowance for RCS flow uncertainty (-2.1 percent)

12. SGTP leveling (0-percent uniform)

    - Maximizes reactor coolant volume and fluid release

    - Maximizes heat transfer area across the SG tubes

    - Reduces coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow

Thus, based on the previously discussed conditions and assumptions, a bounding analysis for the HNP was made for the release of M&E from the RCS in the event of a LOCA at core power of 2900 MWt.

6.4.1.2 Description of Analyses and Evaluations

The evaluation model used for the long-term LOCA M&E release calculations is the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the Nuclear Regulatory Commission (NRC). The approval letter is included with Reference 1. This methodology has previously been applied to the HNP (Reference 2).

6.4.1.2.1 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA M&E analysis, 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 and containment reach an equilibrium state.

2. Refill – the period of time when the lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment M&E releases, it is assumed that 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 M&E to containment. Thus, the refill period is conservatively neglected in the M&E release calculation.

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 phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two-phase.

6.4.1.2.2 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of M&E release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA M&E releases for the HNP SGR/Uprating program. These codes have been used for this analysis since the original plant licensing.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, M&E flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model, discussed in subsection 6.4.1.4.2.

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous M&E release tables and M&E balance tables with data at critical times.

6.4.1.2.3 Break Size and Location

Generic studies (Reference 1, Section 3) have been performed with respect to the effect of postulated break size on the LOCA M&E releases. The double-ended guillotine break (DEGB) has been found to be limiting due to larger mass flowrates 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 for any release purposes:

1. Hot leg (between vessel and steam generator)

2. Cold leg (between pump and vessel)

3. Pump suction (between steam generator and pump)

The break locations analyzed for this program are the DEPS rupture (10.48 ft^2) and the DEHL rupture (9.18 ft^2). Break M&E releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion of each break location.

- The DEHL rupture has been shown in previous studies (Reference 1, Section 3.1) to result in the highest blowdown M&E release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of fluid that exits the core vents directly to containment, bypassing the steam generators. As a result, the reflood M&E 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, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the M&E releases for the hot-leg break blowdown phase are calculated and presented in subsection 6.4.1.4 of this report.

- The cold-leg break location has also been found in previous studies (Reference 1, Section 3.1) to be much less limiting in terms of the overall containment energy releases. 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. During reflood, the flooding rate is greatly reduced, and the energy release rate into the containment is reduced. Therefore, the cold-leg break is bounded by other breaks and no further evaluation is necessary.

- The pump suction break combines the effects of the relatively high core-flooding rate, as in the hot-leg break, and the additional 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 RCS in calculating the releases to containment.

6.4.1.2.4 Application of Single-Failure Criterion

An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period, which is limited by the DEHL break, since the combination of signal delay plus diesel delay and additional delays in starting the ECCS pumps results in an SI delivery time after the end of blowdown.

Generally, two cases are analyzed to assess the effects of a single failure. The first case assumes minimum safeguards SI flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The other case assumes maximum safeguards SI flow based on no postulated failures that would impact the amount of ECCS flow.

6.4.1.3 Acceptance Criteria

A large LOCA is classified as an ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan Section 6.2.1.3, the relevant requirements are as follows:

- 10 CFR 50, Appendix A

- 10 CFR 50, Appendix K, paragraph I.A

In order to meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

6.4.1.4 Results

6.4.1.4.1 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform, and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 6.4.1-4 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot-leg break M&E release tables, break path 1 refers to the M&E exiting from the reactor vessel side of the break; and break path 2 refers to the M&E exiting from the steam generator side of the break.

Table 6.4.1-5 presents the calculated M&E releases for the blowdown phase of the DEPS break with either minimum or maximum ECCS flows. For the pump suction breaks, break path 1 in the M&E release tables refers to the M&E exiting from the steam generator side of the break; break path 2 refers to the M&E exiting from the pump side of the break.

6.4.1.4.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models—one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena, such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator releases are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flowrates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 M&E release evaluation model in recent analyses, for example, D. C. Cook Docket (Reference 3). Even though the Reference 1 model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). Moreover, this assumption is supported by test data and is further discussed 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 ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that needs to be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data, generated in 1/3-scale tests (Reference 4), are the largest scale data available and thus, most clearly simulate the flow regimes and gravitational effects that would occur in a Pressurized Water Reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate 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.

Additionally, the following justification is also noted. The double-ended pump suction break results in the highest containment pressure post-blowdown. For this break, there are two flowpaths 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 that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam that is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. Descriptions of the test and test results are contained in References 1 and 3.

Table 6.4.1-6 presents the calculated M&E release for the reflood phase of the pump suction double-ended rupture with minimum safeguards. Table 6.4.1-9 presents the calculated M&E

release for the reflood phase of the pump suction double-ended rupture with maximum safeguards.

The transient responses of the principal parameters during reflood are given in Table 6.4.1-7 for the DEPS minimum safeguards case. The transient responses of the principal parameters during reflood are given in Table 6.4.1-10 for the DEPS maximum safeguards case.

6.4.1.4.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 5) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The M&E releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken-loop and intact-loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure. After this point, the EPITOME code completes the SG depressurization. (See subsection 6.4.1.4.5 for additional information.)

The methodology for the use of this model is described in Reference 1. The M&E release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the M&E release available to containment is generated directly from core boil-off/decay heat.

Table 6.4.1-8 presents the two-phase post-reflood M&E release data for the pump suction double-ended case minimum safeguards case. Table 6.4.1-11 presents the two-phase post-reflood M&E release data for the pump suction double-ended maximum safeguards case.

6.4.1.4.4 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS Standard 5.1 (Reference 6) for the determination of decay heat. This standard was used in the M&E release. Table 6.4.1 -12 lists the decay heat curve used in the M&E release analysis, post blowdown, for the HNP SGR/Uprating program.

Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E releases analysis include the following:

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 of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Equation 11 of Reference 6, up to 10,000 seconds and from Table 10 of Reference 6, beyond 10,000 seconds.

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The number of atoms of U-239 produced per second has been assumed to be equal to 70 percent of the fission rate.

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

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model (Reference 1), use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of M&E releases to the containment following a LOCA.

6.4.1.4.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for SG cooldown removing steam generator secondary energy at different rates (i.e., first and second stage rates). The first stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken-loop and intact-loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature, and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. The intermediate equilibrium pressures are chosen as discussed in Reference 1, Sections 2.3 and 3.3. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology (Reference 1), all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F.

6.4.1.4.6 Sources of Mass and Energy

The sources of mass considered in the LOCA M&E release analysis are given inTables 6.4.1-13, and 6.4.1-14 and 6.4.1-15. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA M&E release analysis are given in Tables 6.4.1-16, 6.4.1-17 and 6.4.1-18. The energy sources are listed below.

- RCS water
- Accumulator water (all three inject)
- Pumped SI water
- Decay heat
- Core stored energy
- RCS metal (includes SG tubes)
- SG metal (includes transition cone, shell, wrapper, and other internals)
- SG secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into, and steam out of, the SG secondary)

The energy reference points are as follows.

- Available energy: 212°F; 14.7 psia
- Total energy content: 32°F; 14.7 psia

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

- Time zero (initial conditions)
- End of blowdown time

- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

In the M&E release data presented, no Zirc-water reaction heat was considered because the clad temperature is assumed not to rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The sequence of events for the LOCA transients are shown in Tables 6.4.1-19 through 6.4.1-21.

6.4.1.5 Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. The results of this analysis were provided for use in the containment integrity analysis. (See BOP Licensing Report.) Further, these analyses performed at NSSS power of 2912.4 MWt bound operation at NSSS power of 2787.4 MWt with Westinghouse Delta 75 replacement steam generators. Operation at NSSS power of 2787.4 MWt is bounded by the NSSS power of 2912.4 MWt analysis due to the higher core-stored energy and increased level of decay heat power during the long-term transient, including effects on steam generator secondary conditions, which results in higher long-term M&E releases.

Table 6.4.1-1 System Parameters Initial Conditions For Thermal Uprate	
Parameters	**Value**
Core Thermal Power (MWt)	2900
Reactor Coolant System Total Flowrate (lbm/sec)	28833.33
Vessel Outlet Temperature (°F)	623.2
Core Inlet Temperature (°F)	554.4
Vessel Average Temperature (°F)	588.8
Initial Steam Generator Steam Pressure (psia)	1011
Steam Generator Design	Δ75
Steam Generator Tube Plugging (percent)	0
Initial Steam Generator Secondary Side Mass (lbm)	135948
Assumed Maximum Containment Backpressure (psia)	59.7
Accumulator	
Water Volume (ft^3) per accumulator	1029.4
N_2 Cover Gas Pressure (psia)	555
Temperature (°F)	130
Safety Injection Delay, total (sec) (from SI Signal)	29.0

Note:

Core Thermal Power, RCS Total Flowrate, RCS Coolant Temperatures, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

Table 6.4.1-2 Safety Injection Flow Minimum Safeguards	
RCS Pressure (psig)	**Total Flow (lbm/sec)**
Injection Mode (Reflood Phase)	
0	629.70
20	586.26
40	538.85
60	485.47
80	423.09
100	344.75
120	230.12
140	70.57
160	70.28
180	69.99
Injection Mode (Post-Reflood Phase)	
45	525.50
Cold Leg Recirculation Mode	
45	470.00

Table 6.4.1-3 Safety Injection Flow Maximum Safeguards	
RCS Pressure (psig)	**Total Flow (lbm/sec)**
Injection Mode (Reflood Phase)	
0	997.50
45	860.85
100	646.48
110	597.37
120	534.19
Injection Mode (Post-Reflood Phase)	
45	860.85
Cold Leg Recirculation Mode	
45	~800

Table 6.4.1-4 Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
0.00000	0.00	0.0	0.00	0.00
0.00104	45868.80	30054200.0	45867.69	30052600.00
0.00216	47552.30	31154100.0	47199.60	30916500.00
0.102	39304.00	26044600.0	28042.90	18333800.00
0.201	37478.50	24779600.0	24280.90	15764100.00
0.302	36093.10	23828900.0	21852.30	13996700.00
0.401	34827.60	22987600.0	20649.40	13017000.00
0.501	34050.39	22462800.0	19856.90	12327000.00
0.602	33959.89	22398000.0	19377.19	11866300.00
0.701	33597.50	22187200.0	19006.90	11507400.00
0.801	32894.80	21780400.0	18720.30	11224700.00
0.902	32500.00	21606200.0	18435.90	10965400.00
1.00	32065.09	21429400.0	18227.50	10767100.00
1.10	31559.40	21210000.0	18062.50	10606700.00
1.20	30994.00	20947500.0	17948.50	10484300.00
1.30	30430.30	20688500.0	17876.30	10392500.00
1.40	29856.00	20423000.0	17849.80	10333100.00
1.50	29225.80	20121400.0	17855.80	10296200.00
1.60	28505.90	19756300.0	17884.59	10275700.00
1.70	27723.59	19341000.0	17922.30	10263800.00
1.80	26936.80	18918500.0	17964.00	10257100.00
1.90	26202.19	18529400.0	18006.09	10253600.00
2.00	25494.90	18153900.0	18044.00	10251000.00
2.10	24774.19	17754400.0	18073.30	10246500.00
2.20	24072.50	17354000.0	18092.30	10239400.00
2.30	23379.90	16945900.0	18098.09	10227400.00
2.40	22718.19	16545199.0	18087.30	10209000.00
2.50	22125.90	16177700.0	18060.50	10184000.00
2.60	21586.00	15827900.0	18014.40	10150300.00
2.70	21127.09	15516800.0	17950.50	10108700.00
2.80	20729.09	15230200.0	17868.90	10059000.00
2.90	20386.30	14967800.0	17768.09	10000300.00
3.00	20100.30	14732100.0	17649.19	9932900.00
3.10	19873.50	14528200.0	17515.50	9858500.00
3.20	19692.69	14347400.0	17363.59	9775100.00
3.30	19557.00	14191500.0	17195.00	9683400.00
3.40	19460.00	14058900.0	17007.69	9582000.00
3.50	19396.19	13946000.0	16797.69	9468900.00

Table 6.4.1-4 (Cont.) Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
3.60	19365.09	13853000.0	16580.09	9352300.000
3.70	19371.19	13783400.0	16360.50	9235200.000
3.80	19406.19	13730100.0	16137.20	9116700.000
3.90	19464.90	13696300.0	15917.00	9000200.000
4.00	19532.90	13672300.0	15685.70	8877900.000
4.20	19738.80	13650900.0	15265.70	8657900.000
4.40	19993.50	13655000.0	14869.70	8451300.000
4.60	20282.09	13692500.0	14314.59	8152400.000
4.80	20610.50	13752700.0	13779.70	7867500.000
5.00	21057.19	13864400.0	13258.29	7592000.000
5.20	12936.50	9460400.0	12775.70	7339500.000
5.40	16074.40	11456900.0	12331.70	7108000.000
5.60	16312.29	11507500.0	11897.00	6880800.000
5.80	16516.59	11535400.0	11527.59	6690300.000
6.00	16730.59	11581900.0	11189.40	6515200.000
6.20	16961.30	11578800.0	10850.40	6337000.000
6.40	17183.00	11604200.0	10548.59	6179000.000
6.60	17299.30	11595600.0	10266.90	6030900.000
6.80	17203.30	11489200.0	10006.40	5893700.000
7.00	17385.30	11508300.0	9758.200	5762500.000
7.20	17605.09	11553900.0	9508.900	5629400.000
7.40	17806.00	11594700.0	9262.299	5497600.000
7.60	18010.40	11641500.0	9017.700	5367100.000
7.80	18240.90	11706900.0	8769.200	5234700.000
8.00	18410.80	11752900.0	8525.200	5105600.000
8.20	18093.80	11523000.0	8276.299	4974500.000
8.40	16382.50	10567800.0	8027.399	4844200.000
8.60	15441.70	10032000.0	7782.500	4717300.000
8.80	15336.09	9939700.0	7540.500	4593400.000
9.00	15305.79	9897600.0	7311.000	4477700.000
9.20	15255.20	9849800.0	7090.299	4367300.000
9.40	15139.79	9768900.0	6878.500	4261700.000
9.60	14873.79	9607300.0	6671.399	4158299.750
9.80	14325.79	9294700.0	6468.000	4057400.000
10.0	13642.59	8911300.0	6269.299	3959500.000
10.2	13160.20	8636500.0	6075.000	3864600.000
10.4	12836.50	8449900.0	5886.799	3774000.000
10.6	12542.00	8283700.0	5703.799	3686800.000

Table 6.4.1-4 (Cont.) Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
10.8	12215.90	8105200.00	5526.600	3603000.000
11.0	11846.09	7906800.00	5356.399	3522700.000
11.2	11438.29	7691400.00	5191.000	3444800.000
11.4	11026.79	7477100.00	5032.200	3370000.000
11.6	10630.70	7275200.00	4878.700	3298400.000
11.8	10133.59	7091500.00	4730.100	3229600.000
12.0	9339.200	6857300.00	4584.299	3162400.000
12.2	8865.500	6706800.00	4443.899	3097700.000
12.4	8414.700	6523200.00	4304.299	3033200.000
12.6	7830.600	6240600.00	4164.399	2968600.000
12.8	7226.100	5943700.00	4025.199	2905300.000
13.0	6764.700	5739800.00	3885.899	2844000.000
13.2	6322.100	5415900.00	3741.600	2781600.000
13.4	5889.100	5080400.00	3581.399	2711600.000
13.6	5413.200	4793700.00	3396.800	2631900.000
13.8	4893.700	4497700.00	3174.100	2543800.000
14.0	4385.600	4205400.00	2914.800	2453500.000
14.2	3924.500	3913700.00	2625.300	2363600.000
14.4	3520.199	3660600.00	2318.399	2276700.000
14.6	3188.000	3443000.00	2016.599	2180300.000
14.8	2931.500	3256300.00	1759.300	2059700.000
15.0	2724.399	3074500.00	1562.500	1901400.000
15.4	2277.699	2665900.00	1300.699	1611600.000
15.6	2050.399	2439800.00	1200.599	1492700.000
15.8	1842.099	2218300.00	1116.099	1391100.000
16.0	1635.900	1986300.00	1033.900	1291300.000
16.2	1459.800	1787800.00	950.4000	1189200.000
16.4	1319.500	1631300.00	882.4000	1106500.000
16.6	1240.099	1547600.00	824.2000	1035600.000
16.8	1164.699	1462300.00	771.7000	971000.000
17.0	1072.099	1349000.00	728.0000	917300.000
17.4	911.9000	1149900.00	658.9000	832000.000
17.6	842.0000	1063000.00	633.0000	800100.000
17.8	745.5000	942400.00	614.2999	777200.000
18.0	655.7999	831100.00	603.2999	763800.000
18.2	577.4000	732400.00	596.0999	755200.000
18.4	497.6000	632700.00	589.0000	746500.000
18.6	414.5000	527600.000	571.9000	724900.0000
18.8	333.7999	425500.000	530.0000	672000.0000

Table 6.4.1-4 (Cont.) Double-Ended Hot-Leg Break Blowdown M&E Releases				
	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
19.0	231.8999	296100.000	476.3999	604600.0000
19.2	125.8000	161000.000	441.2999	561000.0000
19.4	0.0000	0.000	421.3999	536200.0000
19.6	0.0000	0.000	344.6000	438300.0000
19.8	0.0000	0.000	248.3999	316800.0000
20.0	0.0000	0.000	146.3999	187300.0000
20.2	0.0000	0.000	0.0000	0.0000

* M&E exiting from theRV side of the break

** M&E exiting from the SG side of the break

Table 6.4.1-5
Double-Ended Pump Suction Break
Blowdown M&E Releases
(Same for all DEPS Runs)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
0.00000	0.0	0.0	0.0	0.0
0.00102	92463.8	51485.9	40808.9	22659.2
0.00209	42572.8	23639.1	42166.4	23411.3
0.101	42230.4	23509.3	21593.0	11977.5
0.201	43036.6	24088.9	24339.8	13513.6
0.301	45457.6	25620.3	24771.9	13767.7
0.501	45811.1	26285.6	22806.4	12697.9
0.701	45947.9	26903.5	21102.6	11756.3
0.902	45086.2	26887.8	20243.6	11286.3
1.20	41912.3	25628.0	19759.7	11019.8
2.00	34787.9	22749.2	19342.1	10781.9
2.30	30697.4	20945.5	18738.3	10443.4
2.40	28263.4	19597.2	18335.2	10217.0
2.50	24063.3	16949.4	17767.9	9900.3
2.60	21095.3	15133.5	17472.9	9737.4
2.80	17157.2	12691.7	16980.6	9466.0
3.00	15029.1	11341.2	16485.0	9194.2
3.20	13842.7	10580.6	16106.8	8989.3
3.50	12676.2	9845.0	15550.9	8688.6
4.00	11360.4	9072.2	14676.1	8218.1
4.40	10583.5	8612.8	14578.8	8184.8
4.60	10270.6	8412.5	15940.2	8956.5
5.20	9807.0	8030.8	15032.2	8473.6
5.60	9623.0	7799.4	14607.6	8252.7
6.00	10323.8	8371.1	14478.0	8182.5
6.40	8675.0	7822.4	13902.4	7842.3
6.60	8565.5	7627.3	13748.1	7754.5
7.00	8889.9	7434.6	13352.2	7524.9
7.80	9919.5	7427.4	12618.0	7095.4
8.40	9731.5	7070.2	12176.3	6837.1
9.80	8255.6	6168.3	11248.9	6300.8
11.0	6936.7	5332.5	10207.0	5715.0
13.2	5686.6	4439.0	8872.5	5013.5
14.0	5240.2	4397.4	7716.0	4524.6
14.6	4334.0	4340.9	6611.3	3819.7
15.0	3377.8	3948.0	5996.9	3142.1
15.4	2633.2	3253.1	5328.1	2578.0
15.8	2068.6	2580.7	4590.7	2104.4
16.2	1673.1	2099.8	3986.9	1736.5

Table 6.4.1-5 (Cont.) Double-Ended Pump Suction Break Blowdown M&E Releases (Same for all DEPS Runs)				
	Break Path No. 1 Flow*		**Break Path No. 2 Flow****	
Time (se)	**(lbm/sec)**	**(Thousand Btu/sec)**	**(lbm/sec)**	**(Thousand Btu/sec)**
16.6	1426.1	1797.1	2580.7	1019.1
16.8	1314.9	1659.6	2272.5	836.3
17.4	967.6	1226.2	2389.1	784.6
17.8	719.0	913.8	3202.4	998.9
18.0	521.7	663.3	3219.9	977.8
18.6	0.0	0.0	1313.0	388.5
19.0	0.0	0.0	234.3	69.8
19.6	0.0	0.0	398.6	121.3
20.0	0.0	0.0	253.5	78.2
21.0	0.0	0.0	237.8	75.3
21.4	0.0	0.0	0.0	0.0

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-6
Double-Ended Pump Suction Break
Minimum Safeguards
Reflood M&E Releases

Time (sec)	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)
21.4	0.0	0.0	0.0	0.0
22.3	0.0	0.0	0.0	0.0
22.5	43.9	51.7	0.0	0.0
22.6	28.1	33.1	0.0	0.0
22.8	35.7	42.1	0.0	0.0
23.3	69.3	81.7	0.0	0.0
25.5	136.7	161.1	0.0	0.0
26.0	147.5	173.9	0.0	0.0
26.5	380.3	450.0	3827.1	514.9
27.6	435.3	515.6	4380.7	609.0
28.6	425.8	504.3	4283.7	599.5
30.6	404.7	479.1	4069.3	576.2
31.6	394.5	467.0	3964.7	564.5
32.7	413.8	489.9	4188.8	583.3
33.7	402.8	476.9	4071.0	572.1
34.7	394.2	466.6	3982.5	562.0
35.7	386.0	456.9	3897.4	552.2
37.7	370.7	438.6	3736.8	533.8
39.7	356.8	422.0	3587.9	516.7
41.7	344.0	406.8	3449.2	500.7
42.7	338.0	399.6	3383.4	493.1
44.7	326.6	386.1	3257.8	478.5
46.7	316.1	373.6	3139.6	464.8
48.7	306.3	361.9	3028.1	451.8
50.7	297.1	351.0	2922.6	439.5
52.7	288.5	340.8	2822.4	427.8
53.7	232.7	274.7	2108.8	351.7
54.8	303.4	358.5	277.4	148.5
60.8	278.4	328.9	268.3	135.9
61.8	274.3	324.0	266.8	133.9
65.8	259.7	306.7	261.6	126.7
69.8	246.2	290.7	256.8	120.2
73.8	233.8	275.9	252.5	114.3
81.8	212.0	250.1	245.1	104.2
82.8	209.5	247.2	244.3	103.1
90.8	192.2	226.7	238.6	95.4
98.8	178.6	210.6	234.2	89.5
108.8	166.1	195.8	230.2	84.2
112.8	162.3	191.3	229.0	82.7

Table 6.4.1-6 (Cont.) Double-Ended Pump Suction Break Minimum Safeguards Reflood M&E Releases				
	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
124.8	154.2	181.8	226.5	79.3
138.8	149.6	176.3	225.0	77.3
152.8	148.1	174.6	224.4	76.6
164.8	148.2	174.7	224.3	76.4
180.8	149.5	176.3	224.6	76.7
184.8	150.9	177.9	225.8	77.3
192.8	153.3	180.7	233.1	79.4
200.8	154.8	182.5	243.6	81.8
206.8	154.9	182.6	252.3	83.4
214.8	153.5	180.9	264.8	85.4
216.6	152.9	180.3	267.8	85.8

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-7
Double-Ended Pump Suction Break, Minimum Safeguards
Principal Parameters During Reflood

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
21.4	223.6	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
22.2	219.0	26.896	0.000	0.80	1.83	0.000	7558.9	7558.9	0.0	99.35
22.3	217.9	28.689	0.000	1.03	1.82	0.000	7521.3	7521.3	0.0	99.35
23.5	216.0	2.751	0.316	1.50	5.91	0.430	7041.5	7041.5	0.0	99.35
24.4	215.7	2.651	0.436	1.63	9.10	0.451	6771.0	6771.0	0.0	99.35
27.6	214.4	4.679	0.654	2.03	15.59	0.684	5293.1	5293.1	0.0	99.35
28.6	214.0	4.409	0.682	2.15	15.59	0.682	5115.4	5115.4	0.0	99.35
31.6	213.0	3.928	0.722	2.45	15.59	0.671	4675.1	4675.1	0.0	99.35
32.7	212.8	4.016	0.731	2.55	15.59	0.683	4915.3	4453.5	0.0	98.75
38.7	212.4	3.544	0.750	3.03	15.59	0.661	4280.7	3804.4	0.0	98.64
45.7	213.1	3.197	0.757	3.51	15.59	0.642	3742.6	3253.6	0.0	98.52
53.7	214.9	2.607	0.756	4.01	15.59	0.571	2519.4	2008.9	0.0	98.06
54.8	215.1	3.063	0.761	4.07	15.44	0.630	493.5	0.0	0.0	92.99
62.8	218.3	2.766	0.761	4.53	14.37	0.622	500.1	0.0	0.0	92.99
71.8	223.4	2.493	0.760	5.00	13.50	0.613	505.7	0.0	0.0	92.99
82.8	230.8	2.225	0.759	5.52	12.85	0.601	510.8	0.0	0.0	92.99
94.8	239.0	2.009	0.759	6.03	12.53	0.587	514.5	0.0	0.0	92.99
108.8	246.7	1.842	0.759	6.57	12.51	0.575	517.1	0.0	0.0	92.99
120.8	252.3	1.755	0.761	7.00	12.69	0.567	518.3	0.0	0.0	92.99
136.8	258.6	1.691	0.765	7.55	13.10	0.562	519.2	0.0	0.0	92.99
150.8	263.4	1.666	0.769	8.00	13.54	0.560	519.5	0.0	0.0	92.99
166.8	268.1	1.654	0.774	8.51	14.09	0.561	519.5	0.0	0.0	92.99
176.8	270.8	1.653	0.778	8.82	14.44	0.562	519.5	0.0	0.0	92.99
182.8	272.3	1.657	0.780	9.00	14.65	0.563	519.4	0.0	0.0	92.99
196.8	275.5	1.671	0.785	9.42	15.08	0.569	518.8	0.0	0.0	92.99
200.8	276.3	1.670	0.787	9.54	15.17	0.571	518.8	0.0	0.0	92.99
216.6	279.4	1.632	0.791	10.00	15.43	0.573	518.9	0.0	0.0	92.99

Table 6.4.1-8
Double-Ended Pump Suction Break
Minimum Safeguards, Post-Reflood M&E Releases

Time (sec)	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)
216.6	148.9	188.1	377.0	107.0
221.6	148.5	187.6	377.4	107.0
301.6	141.5	178.8	384.4	106.9
421.6	130.0	164.3	395.9	106.8
456.6	127.8	161.4	398.1	106.5
496.6	122.5	154.8	403.4	106.9
521.6	120.9	152.8	404.9	106.6
561.6	117.2	148.0	408.7	106.6
566.6	118.1	149.3	407.8	106.2
581.6	117.1	148.0	408.7	106.1
586.6	118.1	149.2	407.8	105.7
606.6	116.8	147.6	409.1	105.5
611.6	117.7	148.8	408.2	105.1
631.6	116.5	147.2	409.4	104.9
636.6	117.4	148.3	408.5	104.5
651.6	116.4	147.1	409.5	104.4
656.6	117.3	148.2	408.6	104.0
671.6	116.3	146.9	409.6	103.9
676.6	117.1	148.0	408.8	103.5
691.6	116.0	146.6	409.8	103.4
711.6	116.8	147.6	409.1	102.6
721.6	116.0	146.6	409.8	102.6
741.6	116.6	147.3	409.3	101.9
766.6	115.5	145.9	410.4	101.5
781.6	116.1	146.7	409.8	100.9
821.6	115.0	145.3	410.9	104.1
831.6	115.7	146.2	410.2	103.6
846.6	114.8	145.0	411.1	103.4
861.6	115.4	145.8	410.5	102.7
946.6	114.3	144.4	411.6	100.2
951.6	65.4	82.6	460.5	112.9
1247.0	65.4	82.6	460.5	112.9
1247.1	71.3	88.9	454.6	110.8
1805.2	71.2	88.9	454.7	110.6
1805.3	64.7	74.5	461.2	52.0
2210.0	61.8	71.1	464.1	52.6
2210.1	61.8	71.1	437.4	78.1
3600.0	54.7	63.0	444.5	79.4
3600.1	43.4	49.9	455.8	68.0
10000.0	31.2	35.9	469.8	65.3
18000.0	26.4	30.4	476.8	60.1
18001.1	26.5	30.5	475.9	62.4

Table 6.4.1-8 (Cont.) Double-Ended Pump Suction Break Minimum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
30000.0	23.4	26.9	480.2	59.6
86400.0	17.1	19.6	487.8	56.6
100000.0	16.3	18.7	488.8	55.7
106400.0	16.0	18.4	489.3	55.3
1000000.0	6.9	7.9	500.5	49.6
2592000.0	4.65	5.35	502.7	49.8
10000000.0	2.2	2.5	505.2	50.06
32000000.0	1.7	1.96	505.6	50.1

Table 6.4.1-9 Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1 Flow*		**Break Path No. 2 Flow****	
Time (sec)	**(lbm/sec)**	**(Thousand Btu/sec)**	**(lbm/sec)**	**(Thousand Btu/sec)**
21.4	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.0	.0	.0	.0	.0
22.2	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.5	43.9	51.7	.0	.0
22.5	32.2	37.9	.0	.0
22.6	28.1	33.1	.0	.0
22.7	30.9	36.4	.0	.0
22.8	35.7	42.1	.0	.0
22.9	43.1	50.8	.0	.0
23.0	49.8	58.7	.0	.0
23.1	56.7	66.7	.0	.0
23.2	62.8	74.0	.0	.0
23.3	69.3	81.7	.0	.0
23.4	73.3	86.4	.0	.0
23.5	77.3	91.1	.0	.0
23.6	81.2	95.6	.0	.0
23.7	84.8	99.9	.0	.0
23.8	88.4	104.1	.0	.0
23.9	91.9	108.2	.0	.0
24.0	95.2	112.2	.0	.0
24.1	98.5	116.0	.0	.0
24.2	101.6	119.7	.0	.0
24.3	104.7	123.4	.0	.0
24.4	107.7	126.9	.0	.0
24.5	110.6	130.3	.0	.0
25.5	136.7	161.1	.0	.0
26.0	147.5	173.9	.0	.0
26.5	380.3	450.0	3827.1	514.9
27.6	435.3	515.6	4380.7	609.0
28.6	425.8	504.3	4283.7	599.5
29.6	415.2	491.6	4176.5	588.0
30.0	410.9	486.6	4133.4	583.3
30.6	404.7	479.1	4069.3	576.2
31.6	394.5	467.0	3964.7	564.5
32.7	434.5	514.6	4406.8	604.6
33.7	421.1	498.7	4269.0	591.6
34.7	412.6	488.5	4182.8	581.8

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
35.6	405.2	479.8	4108.1	573.2
35.7	404.4	478.8	4099.9	572.2
36.7	396.7	469.5	4020.0	563.1
37.7	389.2	460.7	3943.2	554.2
38.7	382.1	452.2	3869.1	545.6
39.7	375.3	444.1	3797.6	537.4
40.7	368.8	436.4	3728.7	529.4
41.7	362.6	428.9	3662.1	521.7
42.2	359.5	425.3	3629.6	517.9
42.7	356.6	421.8	3597.7	514.2
43.7	350.8	414.9	3535.4	507.0
44.7	345.2	408.3	3475.0	500.0
45.7	339.9	401.9	3416.5	493.2
46.7	334.7	395.7	3359.7	486.6
47.7	329.7	389.8	3304.5	480.1
48.7	324.9	384.1	3250.9	473.9
49.7	320.2	378.5	3198.7	467.8
50.7	315.7	373.1	3147.9	461.8
51.7	311.3	367.9	3098.4	456.0
52.7	307.1	362.9	3050.2	450.4
53.7	303.0	358.0	3003.1	444.8
54.7	205.5	242.4	500.9	141.9
55.7	208.9	246.5	494.1	141.2
56.7	207.5	244.8	496.7	141.4
57.7	206.0	243.1	499.3	141.7
58.6	204.7	241.5	501.7	141.9
58.7	204.5	241.3	502.0	142.0
59.7	203.0	239.5	504.7	142.3
60.7	201.5	237.7	507.4	142.6
61.7	199.9	235.8	510.2	142.9
62.7	198.3	233.9	513.0	143.2
63.7	196.7	232.0	515.8	143.5
64.7	195.1	230.1	518.6	143.8
65.7	193.4	228.1	521.5	144.2
66.7	191.7	226.1	524.5	144.5
67.7	190.0	224.1	527.4	144.9
68.7	188.3	222.0	530.5	145.2
69.7	186.5	219.9	533.5	145.6
70.7	184.6	217.7	536.6	146.0

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
71.7	182.8	215.6	539.8	146.4
72.7	180.9	213.3	543.0	146.8
73.7	179.0	211.0	546.3	147.2
74.7	177.0	208.7	549.6	147.7
75.7	175.0	206.3	553.0	148.1
76.7	172.9	203.9	556.4	148.6
77.7	170.8	201.4	559.9	149.1
78.7	168.7	198.9	563.5	149.6
79.7	166.5	196.3	567.1	150.1
80.4	164.9	194.4	569.7	150.5
80.7	164.2	193.6	570.8	150.6
81.7	161.9	190.9	574.7	151.2
82.7	159.5	188.0	578.6	151.8
84.7	157.8	186.0	581.8	151.8
86.7	157.2	185.3	583.2	151.6
88.7	156.6	184.6	584.7	151.3
90.7	156.0	183.9	586.1	151.1
92.7	155.5	183.2	587.5	150.9
94.7	154.9	182.6	588.9	150.6
96.7	154.3	181.9	590.4	150.4
98.7	153.7	181.2	591.8	150.1
100.7	153.1	180.5	593.2	149.9
102.7	152.5	179.8	594.6	149.6
104.7	151.9	179.1	596.0	149.4
105.4	151.7	178.9	596.4	149.3
106.7	151.3	178.4	597.3	149.1
108.7	150.8	177.7	598.7	148.9
110.7	150.2	177.0	600.1	148.6
112.7	149.6	176.3	601.5	148.4
114.7	148.9	175.6	602.9	148.1
114.7	148.9	175.6	602.9	148.1
116.7	148.3	174.8	604.3	147.8
118.7	147.7	174.1	605.6	147.6
120.7	147.1	173.4	607.0	147.3
122.7	146.5	172.7	608.4	147.1
124.7	145.9	172.0	609.7	146.8
126.7	145.3	171.2	611.1	146.5
128.7	144.7	170.5	612.4	146.3
130.7	144.1	169.8	613.8	146.0
132.7	143.4	169.1	615.1	145.8

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
132.8	143.4	169.0	615.2	145.7
134.7	142.8	168.3	616.5	145.5
136.7	142.2	167.6	617.8	145.2
138.7	141.6	166.9	619.2	145.0
140.7	141.0	166.1	620.5	144.7
142.7	140.3	165.4	621.9	144.4
144.7	139.7	164.7	623.2	144.2
146.7	139.1	163.9	624.5	143.9
148.7	138.5	163.2	625.9	143.6
150.7	137.8	162.4	627.2	143.4
152.7	137.2	161.7	628.5	143.1
154.7	136.5	160.9	629.9	142.8
156.7	135.9	160.2	631.2	142.6
158.7	135.3	159.4	632.5	142.3
160.7	134.6	158.6	633.9	142.0
162.7	134.0	157.9	635.2	141.8
163.0	133.9	157.8	635.4	141.7
164.7	133.3	157.1	636.5	141.5
166.7	132.7	156.4	637.8	141.2
168.7	132.0	155.6	639.2	141.0
170.7	131.4	154.8	640.5	140.7
172.7	130.7	154.1	641.8	140.4
174.7	130.1	153.3	643.1	140.2
176.7	129.4	152.5	644.4	139.9
178.7	128.8	151.8	645.7	139.7
180.7	128.1	151.0	647.1	139.4
182.7	127.5	150.2	648.4	139.1
184.7	126.8	149.4	649.7	138.9
186.7	126.2	148.7	651.0	138.6
188.7	125.5	147.9	652.3	138.3
190.7	124.9	147.1	653.6	138.1
192.7	124.2	146.4	654.9	137.8
194.7	123.6	145.6	656.2	137.6
196.7	122.9	144.8	657.5	137.3
197.0	122.8	144.7	657.7	137.3
198.7	122.2	144.0	658.8	137.1
200.7	121.6	143.3	660.1	136.8
202.7	120.9	142.5	661.4	136.5
204.7	120.3	141.7	662.7	136.3
206.7	119.6	140.9	664.0	136.0

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		**Break Path No. 2****	
Time (sec)	**Flow (lbm/sec)**	**Energy (Thousand Btu/sec)**	**Flow (lbm/sec)**	**Energy (Thousand Btu/sec)**
208.7	118.9	140.1	665.3	135.8
210.7	118.3	139.3	666.6	135.5
212.7	117.6	138.5	667.9	135.3
214.7	116.9	137.8	669.2	135.0
216.7	116.2	137.0	670.5	134.8
218.7	115.6	136.2	671.7	134.5
220.7	114.9	135.4	673.0	134.3
222.7	114.2	134.6	674.3	134.0
223.8	113.9	134.2	675.0	133.9

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-10 Double-Ended Pump Suction Break, Maximum Safeguards Principal Parameters During Reflood											
	Flooding						Total	Injection Accum	Spill	
Time (sec)	Temp (°F)	Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	(lbm/sec)			Enthalpy (Btu/lbm)
21.4	223.6	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.2	219.0	26.896	.000	.80	1.83	.000	7558.9	7558.9	.0	99.35
22.3	217.9	28.689	.000	1.03	1.82	.000	7521.3	7521.3	.0	99.35
23.5	216.0	2.751	.316	1.50	5.91	.430	7041.5	7041.5	.0	99.35
24.4	215.7	2.651	.436	1.63	9.10	.451	6771.0	6771.0	.0	99.35
27.6	214.4	4.679	.654	2.03	15.59	.684	5293.1	5293.1	.0	99.35
28.6	214.0	4.409	.682	2.15	15.59	.682	5115.4	5115.4	.0	99.35
31.6	213.0	3.928	.722	2.45	15.59	.671	4675.1	4675.1	.0	99.35
32.7	212.8	4.153	.732	2.55	15.59	.692	5163.7	4396.8	.0	98.40
38.7	212.3	3.671	.751	3.05	15.59	.670	4515.8	3732.7	.0	98.25
45.7	212.9	3.323	.758	3.55	15.59	.653	3988.1	3187.7	.0	98.07
53.7	214.6	3.037	.760	4.05	15.59	.635	3514.6	2700.3	.0	97.88
54.7	214.8	2.427	.754	4.11	15.59	.547	848.2	.0	.0	92.99
55.7	215.1	2.439	.754	4.16	15.59	.550	846.4	.0	.0	92.99
62.7	217.8	2.348	.755	4.50	15.59	.541	849.0	.0	.0	92.99
73.7	223.5	2.190	.756	5.01	15.59	.523	853.6	.0	.0	92.99
86.7	231.6	2.014	.756	5.56	15.59	.500	858.3	.0	.0	92.99
98.7	239.2	1.957	.760	6.04	15.59	.501	858.3	.0	.0	92.99
110.7	245.9	1.901	.763	6.50	15.59	.501	858.3	.0	.0	92.99
124.7	252.6	1.836	.766	7.02	15.59	.502	858.3	.0	.0	92.99
138.7	258.2	1.772	.769	7.51	15.59	.503	858.3	.0	.0	92.99
154.7	263.8	1.699	.772	8.04	15.59	.504	858.4	.0	.0	92.99
170.7	268.5	1.626	.776	8.54	15.59	.504	858.5	.0	.0	92.99
186.7	272.4	1.554	.779	9.01	15.59	.505	858.6	.0	.0	92.99
204.7	276.3	1.474	.782	9.51	15.59	.505	858.7	.0	.0	92.99
223.8	279.7	1.389	.786	10.00	15.59	.505	858.9	.0	.0	92.99

Table 6.4.1-11 Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
223.8	125.6	159.2	735.9	142.8
228.8	125.2	158.8	736.2	142.8
233.8	124.9	158.4	736.6	142.8
238.8	126.2	160.0	735.2	142.3
243.8	125.9	159.6	735.6	142.3
248.8	125.5	159.2	735.9	142.2
253.8	125.2	158.7	736.3	142.2
258.8	124.9	158.3	736.6	142.2
263.8	124.5	157.9	737.0	142.1
268.8	124.2	157.4	737.3	142.1
273.8	125.5	159.1	736.0	141.6
278.8	125.1	158.6	736.3	141.6
283.8	124.8	158.2	736.7	141.6
288.8	124.4	157.8	737.0	141.5
293.8	124.1	157.3	737.4	141.5
298.8	123.7	156.9	737.7	141.5
303.8	125.0	158.5	736.5	141.0
308.8	124.6	158.0	736.8	141.0
313.8	124.3	157.6	737.2	140.9
318.8	123.9	157.1	737.5	140.9
323.8	123.6	156.7	737.9	140.9
328.8	124.8	158.3	736.6	140.4
333.8	124.5	157.8	737.0	140.4
338.8	124.1	157.4	737.4	140.4
343.8	123.7	156.9	737.7	140.3
348.8	123.4	156.4	738.1	140.3
353.8	123.0	156.0	738.4	140.3
358.8	124.2	157.5	737.2	139.8
363.8	123.9	157.1	737.6	139.8
368.8	123.5	156.6	738.0	139.8
373.8	123.1	156.1	738.3	139.7
378.8	122.8	155.7	738.7	139.7
383.8	122.4	155.2	739.1	139.7
388.8	123.6	156.7	737.9	139.2
393.8	123.2	156.2	738.2	139.2
398.8	122.8	155.8	738.6	139.2
403.8	122.6	155.4	738.9	139.1
408.8	122.3	155.1	739.2	139.1
413.8	122.1	154.8	739.4	139.0
418.8	123.4	156.4	738.1	138.5

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
423.8	123.1	156.1	738.4	138.5
428.8	122.8	155.7	738.6	138.4
433.8	122.6	155.4	738.9	138.4
438.8	122.3	155.1	739.2	138.3
443.8	122.0	154.7	739.4	138.3
448.8	121.8	154.4	739.7	138.2
453.8	123.1	156.0	738.4	137.8
458.8	122.8	155.7	738.7	137.7
463.8	122.5	155.3	738.9	137.7
468.8	122.2	155.0	739.2	137.6
473.8	122.0	154.6	739.5	137.6
478.8	121.7	154.3	739.8	137.5
483.8	121.4	154.0	740.0	137.4
488.8	122.7	155.5	738.8	137.0
493.8	122.4	155.2	739.1	136.9
498.8	122.1	154.8	739.4	136.9
503.8	121.8	154.4	739.6	136.8
508.8	121.5	154.1	739.9	136.8
513.8	121.2	153.7	740.2	136.7
518.8	122.4	155.3	739.0	136.3
523.8	122.2	154.9	739.3	136.2
528.8	121.9	154.5	739.6	136.2
533.8	121.6	154.1	739.9	136.1
538.8	121.3	153.8	740.2	136.1
543.8	121.0	153.4	740.5	136.0
548.8	122.1	154.9	739.3	135.6
553.8	121.8	154.5	739.6	135.6
558.8	121.5	154.1	739.9	135.5
563.8	121.2	153.7	740.2	135.5
568.8	120.9	153.3	740.5	135.4
573.8	120.6	153.0	740.8	135.4
578.8	121.8	154.4	739.7	134.9
583.8	121.4	154.0	740.0	134.9
588.8	121.1	153.6	740.3	134.8
593.8	120.8	153.2	740.7	134.8
598.8	120.5	152.8	741.0	134.7
603.8	121.6	154.2	739.8	134.3
608.8	121.3	153.8	740.1	134.3
613.8	121.0	153.5	740.4	134.2
618.8	120.8	153.1	740.7	134.2

Table 6.4.1-11 (Cont.)
Double-Ended Pump Suction Break
Maximum Safeguards, Post-Reflood M&E Releases

	Break Path No. 1*		Break Path No. 2**	
Time (sec)	**Flow (lbm/sec)**	**Energy (Thousand Btu/sec)**	**Flow (lbm/sec)**	**Energy (Thousand Btu/sec)**
623.8	120.5	152.8	741.0	134.1
628.8	120.2	152.4	741.3	134.0
633.8	121.3	153.8	740.2	133.6
638.8	121.0	153.4	740.5	133.6
643.8	120.7	153.0	740.8	133.5
648.8	120.4	152.7	741.1	133.5
653.8	120.1	152.3	741.4	133.4
658.8	121.1	153.6	740.3	133.0
663.8	120.8	153.2	740.6	133.0
668.8	120.5	152.8	740.9	132.9
673.8	120.2	152.4	741.2	137.1
678.8	119.9	152.0	741.6	137.0
683.8	120.9	153.3	740.6	136.6
688.8	120.6	152.9	740.9	136.6
693.8	120.3	152.5	741.2	136.5
698.8	119.9	152.1	741.5	136.4
703.8	119.6	151.6	741.9	136.4
708.8	120.6	152.9	740.9	136.0
713.8	120.2	152.4	741.2	135.9
718.8	119.9	152.0	741.6	135.9
723.8	119.5	151.5	741.9	135.8
728.8	120.4	152.7	741.0	135.4
733.8	120.1	152.3	741.4	135.3
738.8	119.7	151.8	741.7	135.3
743.8	119.4	151.3	742.1	135.2
748.8	120.2	152.5	741.2	134.9
753.8	119.9	152.0	741.6	134.8
758.8	119.5	151.5	742.0	134.8
763.8	120.3	152.6	741.1	134.4
768.8	119.9	152.1	741.5	134.3
773.8	119.5	151.6	741.9	134.3
778.8	119.1	151.1	742.3	134.2
783.8	119.9	152.1	741.5	133.9
788.8	119.5	151.5	741.9	133.8
793.8	119.1	151.0	742.4	133.8
798.8	119.9	152.0	741.6	133.4
803.8	119.4	151.4	742.0	133.4
808.8	119.0	150.9	742.4	133.3
813.8	119.7	151.8	741.7	133.0
818.8	119.3	151.3	742.1	133.0
823.8	118.9	150.7	742.6	132.9

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
828.8	119.6	151.6	741.9	132.6
833.8	119.1	151.0	742.4	132.6
838.8	118.6	150.4	742.8	132.5
843.8	119.3	151.2	742.2	132.2
848.8	118.8	150.6	742.7	132.2
853.8	119.4	151.4	742.1	131.8
858.8	118.9	150.7	742.6	131.8
863.8	119.4	151.4	742.0	131.5
868.8	118.9	150.8	742.6	131.5
873.8	119.4	151.4	742.0	131.2
878.8	118.9	150.7	742.6	131.2
883.8	119.3	151.3	742.1	130.9
888.8	118.7	150.6	742.7	130.9
893.8	119.2	151.1	742.3	130.6
898.8	118.5	150.3	742.9	130.6
903.8	118.9	150.8	742.6	130.4
908.8	118.3	149.9	743.2	130.4
913.8	118.6	150.3	742.9	130.2
918.8	118.8	150.7	742.6	133.8
923.8	119.1	151.0	742.4	133.6
928.8	118.3	150.0	743.1	133.6
933.8	118.5	150.2	743.0	133.3
938.8	118.6	150.4	742.8	133.1
943.8	118.7	150.5	742.8	132.9
948.8	118.7	150.5	742.8	132.7
953.8	118.7	150.5	742.8	132.6
958.8	118.6	150.4	742.9	132.4
963.8	118.5	150.2	743.0	132.3
968.8	118.3	149.9	743.2	132.1
973.8	118.0	149.6	743.5	132.0
978.8	118.4	150.2	743.0	131.7
983.8	118.0	149.6	743.5	131.7
988.8	118.2	149.9	743.2	131.4
993.8	118.3	150.1	743.1	131.2
998.8	118.3	150.0	743.2	131.0
1003.8	118.1	149.7	743.4	130.9
1008.8	117.7	149.3	743.7	130.8
1013.8	117.8	149.4	743.6	130.6
1018.8	64.3	81.5	797.2	144.4
1198.8	61.9	78.5	799.5	144.4
1200.0	61.9	78.5	735.1	183.1

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
1283.4	61.9	78.5	735.1	183.1
1283.5	71.0	88.8	726.0	180.3
1285.0	70.9	88.8	726.0	180.3
1786.3	70.9	88.8	726.0	180.3
1786.4	65.0	74.8	731.9	121.8
3599.9	54.9	63.1	742.1	123.6
3600.1	43.4	49.9	753.6	112.4
18000.1	26.4	30.4	776.7	97.9
18001.1	26.5	30.5	775.3	101.6
30000.1	23.4	26.9	780.3	96.8
106400.1	16.0	18.4	790.4	89.4
1000000.1	6.9	7.9	802.8	79.6
2592000.1	4.7	5.4	805.1	79.5
10000000.0	2.2	2.5	807.6	79.7

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-12
LOCA M&E Release Analysis
Core Decay Heat Fraction

Time (sec)	Decay Heat Generation Rate (Btu/hr)
10	0.052293
15	0.049034
20	0.047562
40	0.041504
60	0.038493
80	0.036410
100	0.034842
150	0.032180
200	0.030432
400	0.026664
600	0.024486
800	0.022943
1000	0.021722
1500	0.019483
2000	0.017903
4000	0.014386
6000	0.012684
8000	0.011645
10000	0.010916
15000	0.010130
20000	0.009368
40000	0.007784
60000	0.006976
80000	0.006439
100000	0.006034
150000	0.005336
200000	0.004859
400000	0.003781
600000	0.003212
800000	0.002844
1000000	0.002589

Table 6.4.1-13
Double-Ended Hot-Leg Break
Mass Balance

		Time (Sec)		
		0.00	20.20	20.20*
		Mass (Thousand lbm)		
Initial	In RCS and ACC	623.83	623.83	623.83
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
Total Available		623.83	623.83	623.83
Distribution	Reactor Coolant	426.13	61.54	61.62
	Accumulator	197.70	147.59	147.52
	Total Contents	623.83	209.13	209.13
Effluent	Break Flow	0.00	414.68	414.68
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	414.68	414.68
Total Accountable**		623.83	623.81	623.81

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

** The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-14 Double-Ended Pump Suction Break Mass Balance Minimum Safeguards								
		Time (Sec)						
		0.00	21.40(1)	21.40(2)	216.6(3)	1247.1(4)	1805.2(5)	3600.0(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	623.83	623.83	623.83	623.83	623.83	623.83	623.83
Added Mass	Pumped Injection	0.00	0.00	0.00	94.23	636.17	929.64	1836.43
	Total Added	0.00	0.00	0.00	94.23	636.17	929.64	1836.43
Total Available		623.83	623.83	623.83	718.06	1260.00	1553.47	2460.26
Distribution	Reactor Coolant	426.13	49.29	56.88	106.51	106.51	106.51	106.51
	Accumulator	197.70	146.24	138.65	0.00	0.00	0.00	0.00
	Total Contents	623.83	195.53	195.53	106.51	106.51	106.51	106.51
Effluent	Break Flow	0.00	428.28	428.28	602.88	1144.82	1438.29	2345.08
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	428.28	428.28	602.88	1144.82	1438.29	2345.08
Total Accountable*		623.83	623.83	623.83	709.39	1251.33	1544.80	2451.59

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

* The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-15
Double-Ended Pump Suction Break
Mass Balance
Maximum Safeguards

		Time (Sec)						
		.00	21.40(1)	21.40(2)	223.8(3)	1283.51(4)	1786.30(5)	3600.00(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	623.83	623.83	623.83	623.83	623.83	623.83	623.83
Added Mass	Pumped Injection	.00	.00	.00	162.65	1070.16	1470.87	2916.35
	Total Added	.00	.00	.00	162.65	1070.16	1470.87	2916.35
Total Available		623.83	623.83	623.83	786.48	1693.99	2094.70	3540.18
Distribution	Reactor Coolant	426.13	49.29	56.88	106.91	106.91	106.91	106.91
	Accumulator	197.70	146.24	138.65	.00	.00	.00	.00
	Total Contents	623.83	195.53	195.53	106.91	106.91	106.91	106.91
Effluent	Break Flow	.00	428.28	428.28	670.90	1578.41	1979.08	3424.56
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	.00	428.28	428.28	670.90	1578.41	1979.08	3424.56
Total Accountable*		623.83	623.81	623.81	777.80	1685.31	2085.99	3531.47

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.

(6) Time at which both SGs equilibrate to 14.7 psia.

* The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-16 Double-Ended Hot-Leg Break Energy Balance				
		Time (Sec)		
		0.00	20.20	20.20*
		Energy (Million Btu)		
Initial Energy	In RCS, Acc, SG	736.00	736.00	736.00
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	5.92	5.92
	Heat From Secondary	0.00	-1.84	-1.84
	Total Added	0.00	4.09	4.09
Total Available		736.00	740.08	740.08
Distribution	Reactor Coolant	254.23	13.12	13.13
	Accumulator	19.64	14.66	14.66
	Core Stored	21.51	9.19	9.19
	Primary Metal	125.65	117.89	117.89
	Secondary Metal	83.41	81.57	81.57
	Steam Generator	231.56	225.19	225.19
	Total Contents	736.00	461.61	461.61
Effluent	Break Flow	0.00	277.98	277.98
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	277.98	277.98
Total Accountable**		736.00	739.59	739.59

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

** The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-17 Double-Ended Pump Suction Break Energy Balance Minimum Safeguards								
		Time (Sec)						
		0.00	21.40$^{(1)}$	21.40$^{(2)}$	216.6$^{(3)}$	1247.13$^{(4)}$	1805.2$^{(5)}$	3600.0$^{(6)}$
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	736.00	736.00	736.00	736.00	736.00	736.00	736.00
Added Energy	Pumped Injection	0.00	0.00	0.00	8.76	59.16	86.45	209.72
	Decay Heat	0.00	5.78	5.78	25.27	96.12	127.31	212.66
	Heat From Secondary	0.00	-.35	-.35	-.35	4.51	6.34	6.34
	Total Added	0.00	5.42	5.42	33.68	159.79	220.10	428.72
Total Available		736.00	741.42	741.42	769.68	895.79	956.09	1164.72
Distribution	Reactor Coolant	254.23	10.77	11.53	27.92	27.92	27.92	27.92
	Accumulator	19.64	14.53	13.77	0.00	0.00	0.00	0.00
	Core Stored	21.51	12.21	12.21	3.91	3.74	3.49	2.71
	Primary Metal	125.65	118.82	118.82	99.71	62.84	52.91	41.33
	Secondary Metal	83.41	83.06	83.06	76.51	50.12	40.37	31.64
	Steam Generator	231.56	230.29	230.29	208.52	134.41	109.90	87.06
	Total Contents	736.00	469.69	469.69	416.56	279.03	234.59	190.66
Effluent	Break Flow	0.00	271.25	271.25	345.42	609.07	706.59	959.89
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	271.25	271.25	345.42	609.07	706.59	959.89
Total Accountable*		736.00	740.93	740.93	761.99	888.10	941.18	1150.55

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

* The difference between total available energy and total accountable energy at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-18
Double-Ended Pump Suction Break
Energy Balance, Maximum Safeguards

		Time (sec)						
		.00	21.40(1)	21.40(2)	223.8(3)	1283.51(4)	1786.30(5)	3600.00(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	736.00	736.00	736.00	736.00	736.00	736.00	736.00
Added Energy	Pumped Injection	.00	.00	.00	15.13	103.25	163.01	378.56
	Decay Heat	.00	5.78	5.78	25.89	98.27	126.30	212.65
	Heat From Secondary	.00	-.35	-.35	-.35	4.65	6.22	6.22
	Total Added	.00	5.42	5.42	40.66	206.17	295.54	597.43
Total Available								
Distribution	Reactor Coolant	736.00	741.42	741.42	776.66	942.16	1031.53	1333.43
	Accumulator	254.23	10.77	11.53	28.01	28.01	28.01	28.01
	Core Stored	21.51	12.21	12.21	3.91	3.74	3.51	2.71
	Primary Metal	125.65	118.82	118.82	99.51	62.17	53.11	41.37
	Secondary Metal	83.41	83.06	83.06	76.67	49.38	40.48	31.68
	Steam Generator	231.56	230.29	230.29	208.93	132.41	110.07	87.03
	Total Contents	736.00	469.69	469.69	417.04	275.70	235.19	190.80
Effluent	Break Flow	.00	271.25	271.25	351.92	658.77	779.73	1127.39
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	271.25	271.25	351.92	658.77	779.73	1127.39
Total Accountable*		736.00	740.93	740.93	768.96	934.47	1014.91	1318.18

Notes to Table 6.4.1-18:

(1) End of Blowdown

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.

(6) Time at which both SGs equilibrate to 14.7 psia.

* The difference between total available energy and total accountable energy at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-19
Double-Ended Hot-Leg Break
Sequence Of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.1	Low Pressurizer Pressure SI Setpoint - 1714.3 psia reached by SATAN
12.7	Broken Loop Accumulator Begins Injecting Water
12.8	Intact Loop Accumulator Begins Injecting Water
20.20	End of Blowdown Phase

Table 6.4.1-20 Double-Ended Pump Suction Break Minimum Safeguards Sequence Of Events	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.4	Low Pressurizer Pressure SI Setpoint - 1714.3 psia reached by SATAN
14.1	Broken Loop Accumulator Begins Injecting Water
14.5	Intact Loop Accumulator Begins Injecting Water
21.4	End of Blowdown Phase
32.4	Safety Injection Begins
53.27	Broken Loop Accumulator Water Injection Ends
53.72	Intact Loop Accumulator Water Injection Ends
216.6	End of Reflood Phase
1247.1	Broken Loop SG Secondary Side pressure reaches the first intermediate pressure
1805.2	Intact Loop SG Secondary Side pressure reaches the second intermediate pressure
2210.0	Cold Leg Recirculation Begins
3600.0	Broken and Intact Loop SG pressure reaches equilibrium at 14.7 psia
3.2E+07	Transient Modeling Terminated

Table 6.4.1-21 Double-Ended Pump Suction Break Maximum Safeguards Sequence Of Events	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.4	Low Pressurizer Pressure SI Setpoint - 1714.3 psia reached by SATAN
14.1	Broken Loop Accumulator Begins Injecting Water
14.5	Intact Loop Accumulator Begins Injecting Water
21.4	End of Blowdown Phase
32.4	Safety Injection Begins
53.87	Broken Loop Accumulator Water Injection Ends
54.32	Intact Loop Accumulator Water Injection Ends
223.8	End of Reflood Phase
1283.5	Broken Loop SG Secondary Side pressure reaches the first intermediate pressure
1786.3	Intact Loop SG Secondary Side pressure reaches the second intermediate pressure.
1200.0	Cold Leg Recirculation Begins
3600.0	Broken and Intact Loop SG pressure reach equilibrium at 14.7 psia
2.592E+06	Transient Modeling Terminated

6.4.2 Short-Term LOCA Mass and Energy Releases

6.4.2.1 Introduction

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. The subcompartments evaluated include the steam generator compartment, the reactor cavity region, and the pressurizer compartment. For the SG compartment and the reactor cavity region, the fact that the HNP is approved for leak-before-break (LBB) was used to qualitatively demonstrate that any changes associated with the SGR/Uprating program are offset by the LBB benefit of using the smaller RCS nozzle breaks. This demonstrates that the current licensing bases for these subcompartments remain bounding. For the pressurizer compartment, the Reference 5 methodology was applied to calculate pressurizer spray line and surge line M&E releases. The results of this evaluation have been provided to CP&L for use in the pressurizer subcompartment evaluation. (See the BOP Licensing Report.)

6.4.2.2 Description of Analyses and Evaluations

The current licensing basis analyses for short-term LOCA mass and energy releases are presented in Section 6.2.1.2 of the FSAR (Reference 7). These mass and energy releases were generated with the Westinghouse 1975 mass and energy model (Reference 5) for the following breaks to support subsequent analyses of the reactor cavity, steam generator, and pressurizer compartments:

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

Mass and energy releases for these breaks are presented in HNP FSAR Tables 6.2-12 through 6.4-18.

A reanalysis was conducted to determine the effect of the SGR/Uprating on the short-term LOCA-related M&E releases that support subcompartment analyses discussed in Chapter 6.2.1.2 of the HNP FSAR (Reference 7). The HNP was licensed for LBB by Reference 8. Therefore, only breaks in the largest branch lines are analyzed, which are the pressurizer surge line and spray line breaks found in the HNP FSAR (cases 6 and 7 from above). The remaining FSAR breaks (i.e., breaks in the main RCS piping, cases 1 through 5) have been eliminated by LBB and therefore, the M&E releases associated with these breaks would bound any RCS primary break considered under the LBB exemption. This evaluation addresses the impact of the SGR/Uprating and other relevant issues on the current licensing basis for the HNP.

The subcompartment analysis is performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) which accompanies a high-energy line pipe rupture within the subcompartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown M&E release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown M&E release rates are affected by the initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions. Short-term blowdown transients are characterized by a peak M&E release rate that occurs during a subcooled condition. Therefore, using lower temperatures, which maximizes the short-term LOCA M&E releases, data representative of the lowest inlet and outlet temperatures (with uncertainty subtracted) were used for the HNP SGR/Uprating analysis.

HNP has a temperature operating range for T_{avg} of 572°F to 588.2°F. For this evaluation, an RCS pressure of 2301 psia (2250 + 51 psi uncertainty), a vessel outlet temperature of 598.2°F, and a vessel/core inlet temperature of 530.6°F were considered for the uprating, which includes consideration of the lower end to the temperature operating range with a temperature uncertainty of -6°F.

Additionally, due to the short time period (0-3 seconds) for which these events are analyzed, the ECCS system is not modeled. Since the ECCS will not start in this short time period, single failures in the ECCS and Engineered Safeguards are not a concern and are not considered.

6.4.2.3 Acceptance Criteria

The NRC's NUREG-0800, Section 6.2.1.3, "Mass and Energy Release analysis for Postulated Loss-of-Coolant Accidents," subsection II, provides guidance on the NRC's expectations for what must be included in a LOCA mass and energy release calculation. The NRC has determined that the Westinghouse M&E models described in WCAP-8264-P-A, Rev. 1 (Reference 5) satisfy those expectations.

6.4.2.5 Results

The results of the pressurizer surge line and spray line break analyses are given inTables 6.4.2-1 and 6.4.2-2. The methodology described in Reference 5 was used.

Per Reference 8, the HNP is approved for LBB. LBB eliminates the dynamic effects of postulated primary loop pipe ruptures from the design basis. This means that the current breaks (a double-ended circumferential rupture of the reactor coolant cold leg, hot leg, and the steam generator inlet nozzle, used for the SG compartments, and a 150 in^2 RV inlet break for the reactor cavity region) no longer have to be considered for the short-term effects. Since the RCS piping has been eliminated from consideration, the large branch nozzles must be considered for design verification. This includes the surge line, accumulator line, and the RHR line. These smaller breaks, which are outside the cavity region, would result in minimal asymmetric pressurization in the reactor cavity region. Additionally, compared to the large RCS double-ended ruptures, the differential loadings are significantly reduced. For example, the peak break compartment pressure can be reduced by a factor of greater than 2, and the peak differential across an adjacent wall can be reduced by a factor of greater than 3, if only the nozzle breaks are considered. Therefore, since the HNP is approved for LBB, the decrease in M&E releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets any increased releases associated with the lower RCS temperatures as a result of the SGR/Uprating. The current licensing basis subcompartment analyses that consider breaks in the RCS remain bounding.

6.4.2.6 Conclusions

The short-term LOCA-related M&E releases discussed in Chapter 6.2 of the HNP FSAR have been reviewed to assess the effects associated with the SGR/Uprating project. New analyses were performed for the pressurizer surge line and spray line breaks. The results of these new analyses appear in Tables 6.4.2-1 and 6.4.2-2. The results of these analyses were provided for use in the pressurizer subcompartment structural analysis. (See the BOP Licensing Report.) Since the HNP is approved for LBB, the decrease in M&E releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets any increased releases associated with the SGR/Uprating project. The current licensing basis subcompartment analyses that consider breaks in the primary loop RCS piping (i.e., steam generator subcompartments and reactor cavity region), therefore, remain bounding. Further, this analysis at NSSS power of 2912.4 MWt bounds operation at NSSS power of 2787.4 MWt with the Model Delta 75 replacement steam generators, since the lower bound RCS temperatures occur at the uprated power of 2912.4 MWt, which results in higher short-term M&E releases.

Table 6.4.2-1 Short-Term Mass and Energy Release for a Double-Ended Pressurizer Surge Line Break			
Time (sec)	**Break Flow (lbm/sec)**	**Break Energy (Btu/sec)**	**Enthalpy (Btu/lbm)**
0.00000	0.0000000E+00	0.0000000E+00	0.00
0.00100	2.1103676E+03	1.3534502E+06	641.33
0.00904	9.4237946E+03	6.0692558E+06	644.04
0.01004	9.8820135E+03	6.3686254E+06	644.47
0.02004	1.3838759E+04	8.9630591E+06	647.68
0.03003	1.4863396E+04	9.6634563E+06	650.15
0.04003	1.6093160E+04	1.0502445E+07	652.60
0.05001	1.6159154E+04	1.0577799E+07	654.60
0.06002	1.6553060E+04	1.0864070E+07	656.32
0.07001	1.6655118E+04	1.0953752E+07	657.68
0.08006	1.6433660E+04	1.0835927E+07	659.37
0.09004	1.6416382E+04	1.0846517E+07	660.71
0.10002	1.6907343E+04	1.1180615E+07	661.29
0.20005	1.4938162E+04	1.0006279E+07	669.85
0.30001	1.4111884E+04	9.4848110E+06	672.12
0.40005	1.4341020E+04	9.6094731E+06	670.07
0.50001	1.4291815E+04	9.5529874E+06	668.42
0.60001	1.4319763E+04	9.5477666E+06	666.75
0.70001	1.4293723E+04	9.5143628E+06	665.63
0.80003	1.4305338E+04	9.5119696E+06	664.92
0.90017	1.4272873E+04	9.4888186E+06	664.81
1.00000	1.4151129E+04	9.4151449E+06	665.33
1.10016	1.3962764E+04	9.3022100E+06	666.22
1.20014	1.3846967E+04	9.2358276E+06	666.99
1.30006	1.3682495E+04	9.1379399E+06	667.86
1.40019	1.3597790E+04	9.0918136E+06	668.62
1.50001	1.3546720E+04	9.0685560E+06	669.43
1.60021	1.3469738E+04	9.0282535E+06	670.26
1.70011	1.3380283E+04	8.9786760E+06	671.04
1.90006	1.3193815E+04	8.8690636E+06	672.21
2.00049	1.3130329E+04	8.8308523E+06	672.55

Note: The tabulated energy releases should be increased 8.35% in order to bound operation at a T_{avg} of 572°F with a -6.0°F temperature uncertainty and at best estimate RCS flow.

Table 6.4.2-2 Short-Term Mass and Energy Release for a Double-Ended Pressurizer Spray Line Break			
Time (sec)	**Mass Flow (lb/sec)**	**Energy Flow (Btu/sec)**	**Avg. Enthalpy (Btu/lb)**
0.00000	0.0000000E+00	0.0000000E+00	0.00
0.01103	5.5518493E+03	3.2905004E+06	592.69
0.02003	5.7362822E+03	3.3924253E+06	591.40
0.03004	5.7848213E+03	3.4191326E+06	591.05
0.04012	5.8059940E+03	3.4305494E+06	590.86
0.05008	5.6819445E+03	3.3609717E+06	591.52
0.06007	5.6565506E+03	3.3467035E+06	591.65
0.07002	5.7317091E+03	3.3884934E+06	591.18
0.08006	5.7132020E+03	3.3780441E+06	591.27
0.09004	5.6174032E+03	3.3245990E+06	591.84
0.10006	5.5866929E+03	3.3074862E+06	592.03
0.20007	5.5623287E+03	3.2933991E+06	592.09
0.30000	5.4512179E+03	3.2308583E+06	592.69
0.40009	5.3230373E+03	3.1592006E+06	593.50
0.50001	5.2181352E+03	3.1007846E+06	594.23
0.60009	5.1229225E+03	3.0478652E+06	594.95
0.70015	5.0239055E+03	2.9929317E+06	595.74
0.80014	4.9453679E+03	2.9493564E+06	596.39
0.90018	4.8895171E+03	2.9183478E+06	596.86
1.00013	4.9513805E+03	2.9521263E+06	596.22
1.10003	4.9736299E+03	2.9641328E+06	595.97
1.20003	5.0045052E+03	2.9809611E+06	595.66
1.30005	5.0311580E+03	2.9955387E+06	595.40
1.40002	5.0433071E+03	3.0021727E+06	595.28
1.50009	5.0585074E+03	3.0105403E+06	595.14
1.60003	5.0654476E+03	3.0143923E+06	595.09
1.70006	5.0667942E+03	3.0151892E+06	595.09
1.80003	5.0675444E+03	3.0156876E+06	595.10
1.90011	5.0613076E+03	3.0123487E+06	595.17
2.00022	5.0532974E+03	3.0080471E+06	595.26

Note: The tabulated energy releases should be increased 0.4% in order to bound operation at a T_{avg} of 572°F with a -6.0°F temperature uncertainty.

6.4.3 References

1. WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Nonproprietary), "Westinghouse LOCA Mass & Energy Release Model for Containment Design - March 1979 Version," May 1983.

2. NUREG-1038, "Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Plant, Units 1 and 2," Section 6.2.1.3, November 1983.

3. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1," June 9, 1989.

4. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

5. WCAP-8264-P-A, Rev. 1 (Proprietary), WCAP-8312-A (Nonproprietary), "Westinghouse Mass and Energy Release Data For Containment Design," August 1975.

6. ANSI/ANS-5.1 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

7. Shearon Harris Nuclear Power Plant Final Safety Analysis Report, Section 6.2.

8. NRC Letter from G. W. Knighton to E. E. Utley (CP&L), "Request for Exemption from a portion of General Design Criteria 4 of Appendix A to 10 CFR Part 50 Regarding the need to Analyze Large Primary Loop Pipe Ruptures as Structural Design Basis for Shearon Harris Nuclear Power Plant Unit 1," June 5, 1985.

6.5 Main Steamline Break Mass and Energy Releases

6.5.1 Main Steamline Break Mass and Energy Releases Inside Containment

6.5.1.1 Introduction

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment and elevated containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. These variations make it difficult to determine the absolute worst cases for either containment pressure or temperature evaluation following a steamline break. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the main steamline break (MSLB) mass and energy (M&E) releases for use in containment integrity analysis.

6.5.1.2 Description of Analyses

The description of the analysis and methods pertaining to the main steamline break mass and energy releases inside containment are presented below.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectra of both variables have been evaluated. At plant power levels of 102 percent, 70 percent, 30 percent and 0 percent of nominal full-load power, two break sizes have been defined. These break areas are defined as the following.

1. A full double-ended guillotine break (DEGB) downstream of the flow restrictor in one steamline. Note that a DEGB is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The full DEGB represents the largest break of the main steamline producing the highest mass flowrate from the faulted-loop steam generator.

2. A small split rupture that will neither generate a steamline isolation signal from the Westinghouse Solid-State Protection System (SSPS) nor result in water entrainment in the break effluent. Reactor protection and safety injection actuation functions are obtained from containment pressure signals.

The 12 cases included in the Harris Nuclear Plant (HNP) SGR/Uprating analysis have been chosen based on the results of the analyses presented in the HNP FSAR, subsection 6.2.1.4. The cases, listed in subsection 6.5.1.2.16 of this licensing report, have been analyzed assuming operation with the Westinghouse-design Model Delta 75 replacement steam generators at the uprated power condition. Other important plant conditions and features are discussed in the following paragraphs.

6.5.1.2.1 Initial Power Level

Steamline breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the steam generators, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during full-power, or near full-power, operation may be greater than for breaks occurring with the plant in a low-power, or hot-shutdown, condition. Additionally, pressure in the steam generators changes with increasing power and has a significant influence on the rate of blowdown.

Because of the opposing effects on mass versus energy release for the MSLB due to a change in initial power level, a single power level cannot be specified as the worst case for either the containment pressure cases or the containment temperature cases. Therefore, representative power levels including 102 percent, 70 percent, 30 percent and 0 percent of nominal full-power conditions have been investigated for HNP as presented in the HNP FSAR, based on the information in Reference 1. Reference 1 has been reviewed and approved by the NRC for use in MSLB analysis inside containment. Additional discussion is provided in subsection 6.5.1.2.16 of this report.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.5.1-1 and 6.5.1-2 identify the values assumed for RCS pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, steam generator pressure, and feedwater enthalpy corresponding to each power level analyzed. Steamline break mass and energy releases assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, there is more mass and energy available for release into containment. The thermal design flowrate has been used for the RCS flow input consistent with the assumptions documented in Reference 1. The thermal design flowrate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the SGR/Uprating program have been applied only to the RCS average temperature (6°F), the steam generator mass (8 percent narrow-range span), and the power fraction (2 percent) and feedwater enthalpy (2°F) at full power. Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the amount of mass or energy discharged into containment.

6.5.1.2.2 Single-Failure Assumptions

To avoid unnecessary conservatism, bounding multiple failure assumptions have not been made in the analysis. Each case analyzed considered only one single failure. One of these failures results in minimum containment spray and fan coolers to allow for a failure of a train of containment safeguards features. The following single failures are postulated (discussed also in Reference 1), which may significantly affect the containment results.

a. Failure of the Main Steam Isolation Valve (MSIV) in the Faulted Loop

The main steamline isolation function is accomplished via the MSIV in each of the three steamlines. Each valve closes on an isolation signal to terminate steam flow from the associated steam generator. The main steamline rupture upstream of this valve, as postulated for the inside-containment analysis, creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSIV successfully closes. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of feedwater and auxiliary feedwater addition are terminated. If the faulted-loop MSIV fails to close, blowdown from more than one steam generator is prevented by the closure of the corresponding MSIV for each intact-loop steam generator. Therefore, there is no failure of a single MSIV that could cause continued blowdown from multiple steam generators.

In addition to the continued blowdown from the faulted-loop steam generator after MSIV closure, the steam in the unisolable section of the steamline needs to be considered. An MSIV failure can impact the mass and energy releases, since a failed MSIV will result in a larger unisolable steamline volume. The analytical method of addressing the steamline piping blowdown and the effect of an MSIV failure is dependent on break type, as discussed in subsection 6.5.1.2.16.

b. Failure of the Main Feedwater Isolation Valve (MFIV) in the Faulted Loop

If the MFIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, backup isolation is provided via the main feedwater flow control valve (MFCV) closure. The additional inventory between the MFIV and the MFCV in the faulted loop would be available to be released to containment. (The effects of this single-failure assumption are being addressed separately in the BOP Licensing Report for the steam generator replacement and power uprate analysis.)

c. Failure of the Emergency Diesel Generator (EDG)

For the circumstance where offsite power is lost, power will be lost to the reactor coolant pumps (RCPs) and the EDGs will be relied upon to supply emergency power to the safeguards equipment. If one EDG fails in this situation, one train of safety injection (SI) as well as one train of the containment safeguards functions will be lost. The only effect on the mass and energy releases is the loss of one SI train, a longer delay until SI actuation, and RCP trip. As noted later, minimum SI flow is assumed for all cases. The effect of reduced containment safeguards is accounted for in the containment response analysis. The assumption of a trip of all the RCPs coincident with reactor trip is less limiting than with offsite power available since the mass and energy releases are reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer. Therefore, all MSLB M&E release cases are analyzed with the RCPs continuing to operate.

6.5.1.2.3 Main Feedwater System

The rapid depressurization that occurs following a steamline rupture typically results in large amounts of water being added to the steam generators through the main feedwater system. Rapid-closing MFIVs or MFCVs in the main feedwater lines limit this effect. The feedwater addition that occurs prior to closing of the MFIVs or MFCVs influences the steam generator blowdown in several ways. First, because the water entering the steam generator is subcooled, it lowers the steam pressure thereby reducing the flowrate out of the break. As the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the control valves will flash into the steam generators providing additional secondary fluid which may exit out of the rupture. Secondly, the increased flow causes an increase in the total heat transfer from the primary to secondary systems resulting in greater integrated energy being released out of the break.

Following the initiation of the MSLB, main feedwater flow is conservatively modeled by assuming that sufficient feedwater flow is provided to match or exceed the steam flow prior to reactor trip. The initial increase in feedwater flow (until fully isolated) is in response to the feedwater control valve opening up in response to the steam flow/feedwater flow mismatch, or the decreasing steam generator water level as well as due to a lower backpressure on the feedwater pump as a result of the depressurizing steam generator. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following the safety injection signal, is assumed to be a total of 10 seconds, accounting for delays associated with signal processing plus MFIV stroke time. (At zero-power initial conditions, the feedwater isolation response time, following the safety injection signal, is assumed to be a total of 12 seconds.) (For the circumstance in which the MFIV in the faulted loop fails to close, the effects of the feedwater isolation response time are being addressed separately in the BOP Licensing Report for the SGR/Uprating program.)

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation valve may flash to steam if the feedwater temperature exceeds the saturation temperature. This unisolable feedwater line volume is an additional source of fluid that can increase the mass discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop.

For the circumstance in which the MFIV in the faulted loop fails to close, the effects of the increase in the unisolable feedwater line volume are being addressed separately in the BOP Licensing Report for the SGR/Uprating program.

Steamline break mass and energy releases assuming a main feedwater temperature at the high end of the feedwater temperature window are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release into containment.

6.5.1.2.4 Auxiliary Feedwater System

Generally, within the first minute following a steamline break, the auxiliary feedwater (AFW) system is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The auxiliary feedwater flow to the faulted and intact steam generators has been assumed to be a constant value in the steamline break analysis inside containment. A high AFW flowrate to the faulted-loop steam generator is conservative for the steamline break event; therefore, these flows have been maximized. Conversely, a low AFW flowrate is conservative for the intact-loop steam generators; thus, these flows have been minimized. The volume of the AFW piping is minimized. Purging of AFW piping is not assumed since a minimum volume permits colder AFW to be injected into the steam generator rather than any hotter auxiliary feedwater resident in the piping. The more dense injected AFW causes a greater mass addition to the faulted-loop steam generator than if the resident auxiliary feedwater had to be purged prior to the flow of AFW into the steam generator. Auxiliary feedwater flow to the faulted-loop steam generator has been assumed up until the time of automatic auxiliary feedwater isolation on a steamline ΔP signal. Flashing of water resident in the AFW piping after isolation will not occur since the fluid temperature is less than the saturation temperature for containment pressure conditions. After isolation, auxiliary feedwater flow continues to intact-loop steam generators.

6.5.1.2.5 Steam Generator Fluid Mass

A maximum initial steam generator mass in the faulted-loop steam generator has been used in all of the analyzed cases. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. The initial mass has been calculated as the value corresponding to the programmed level +8 percent narrow-range span. Minimum initial masses in the intact-loop steam generators have been used in all of the analyzed cases. The use of reduced initial steam generator masses minimizes the availability of the heat sink afforded by the steam generators on those loops. The initial masses have been calculated as the value corresponding to the programmed level -8 percent narrow-range span. Steam generator reverse heat transfer is discussed in the following subsection.

All steam generator fluid masses are calculated corresponding to 0 percent tube plugging, which is conservative with respect to maximizing the fluid inventory and the primary-to-secondary heat transfer through the faulted-loop steam generator resulting from the steamline break.

6.5.1.2.6 Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops become sources of energy that can be transferred to the steam generator with the broken steamline. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant.

This energy is then available to be transferred to the steam generator with the broken steamline. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.5.1.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an f(l/ D) = 0 is used]. This is consistent with the expectations of the NRC as presented in Section 6.2.1.4 of the Standard Review Plan. For the HNP analysis, no entrainment is assumed in the break effluent. The assumption of saturated steam being released for all break types is a conservative assumption that maximizes the energy release into containment.

6.5.1.2.8 Steamline Volume Blowdown

The contribution to the mass and energy releases from the steam in the secondary plant main steam loop piping and header has been included in the mass and energy release calculations. The initial flowrate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. This blowdown is calculated only for the DEGB steamline break.

For the split-rupture steamline break, the unisolable steam mass in the piping is included as part of the initial inventory in the faulted-loop steam generator since the break is not large enough to cause a sudden decompression of the piping.

The analytical method of addressing the steamline piping blowdown and the effect of an MSIV failure or no MSIV failure is discussed in subsection 6.5.1.2.16.

6.5.1.2.9 Main Steamline Isolation

Steamline isolation is assumed in all three loops to terminate the blowdown from the two intact steam generators. A delay time of 7 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed.

6.5.1.2.10 Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, safety injection, steamline isolation, and feedwater isolation. (Analysis of the containment response to the MSLB is presented in the BOP Licensing Report.) The protection system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 6.5.1-3. The setpoints used are conservative values with respect to the plant-specific values delineated in the HNP Technical Specifications for Uprate conditions.

For the double-ended rupture MSLB at all power levels, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in any loop that initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal.

For the split-rupture steamline breaks at all power levels, no mitigation signals are received from either the Reactor Protection System or any secondary-side signals produced by the Engineered Safety Features Actuation System. The first protection system signal actuated is assumed to be the High Containment Pressure which initiates safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal. Steamline isolation is initiated following receipt of the High-High Containment Pressure signal.

The turbine stop valve is assumed to close instantly following the reactor trip signal.

6.5.1.2.11 Safety Injection System

Minimum safety injection system (SIS) flowrates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steamline break inside containment since the mass and energy releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.5.1.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant could drop below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat, however, does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.5.1.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 2). The existing analysis assumed the use of the 1971 standard (+20 percent uncertainty) for the decay heat. The assumption of using the 1979 version represents a deviation from the current licensing-basis analysis for HNP. This version of the decay heat input has been applied previously to other HNP licensing-basis safety analyses (FSAR Chapter 15, Reference 15.0.10-3).

6.5.1.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steamline break analyses.

6.5.1.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions, including hot zero power (HZP) stuck-rod moderator density coefficients, are used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.5.1.2.16 Description of Analysis Cases

The system transient that provides the break flows and enthalpies of the steam release through the steamline break inside containment has been analyzed with the LOFTRAN (Reference 3) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick-metal heat storage including steam generator thick-metal mass and tubing, and reverse steam generator heat transfer. As noted in Section 6.5.1.2.7, no entrainment is assumed in the break effluent for the HNP analysis. The assumption of saturated steam being released for all break types is a conservative assumption that maximizes the energy release into containment. This does not reflect a deviation in the HNP licensing basis since the current MSLB mass and energy releases analysis does not assume entrainment.

The existing MSLB M&E analysis inside containment was performed using the MARVEL code as documented in WCAP-8822. The use of the LOFTRAN code for the analysis of the MSLB M&E releases is documented in Supplement 1 of WCAP-8822 (Reference 1) and has been reviewed and approved by the NRC for this application. The LOFTRAN code has been utilized previously for the HNP licensing-basis safety analyses.

The HNP NSSS has been analyzed to determine the transient steam mass and energy releases inside containment following a steamline break event. Superheated steam is not assumed in the mass and energy releases from the faulted-loop steam generator for this analysis. The approved methodology for the MSLB M&E releases inside containment as documented in Supplement 2 of WCAP-8822 (Reference 1) does not require that the effects of steam superheat be considered for input to an analysis assuming a large, dry containment. Since the HNP containment design is of this type, the steam superheat assumption has not been considered in this analysis for the steam generator replacement and power uprate program. The resulting tables of mass and energy releases are used as input conditions to the analysis of the containment response.

The following licensing-basis cases of the MSLB inside containment have been analyzed at the noted conditions for the SGR/Uprating program.

Case 1: Full double-ended (1.4 ft^2) rupture at 102 percent power

Case 2: 0.687 ft^2 split rupture at 102 percent power – no single failure in MSLB transient

Case 3: 0.687 ft^2 split rupture at 102 percent power – MSIV failure

Case 4: Full double-ended (1.4 ft^2) rupture at 70 percent power

Case 5: 0.675 ft^2 split rupture at 70 percent power – no single failure in MSLB transient

Case 6: 0.675 ft^2 split rupture at 70 percent power – MSIV failure

Case 7: Full double-ended (1.4 ft^2) rupture at 30 percent power

Case 8: 0.666 ft^2 split rupture at 30 percent power – no single failure in MSLB transient

Case 9: 0.666 ft^2 split rupture at 30 percent power – MSIV failure

Case 10: Full double-ended (1.4 ft^2) rupture at hot standby (0 percent power)

Case 11: 0.558 ft^2 split rupture at 0 percent power – no single failure in MSLB transient

Case 12: 0.558 ft^2 split rupture at 0 percent power – MSIV failure

For the double-ended rupture cases, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the integral flow restrictor area of 1.4 ft^2, which is less than the actual area of 4.9 ft^2 for the main steam piping inside containment. The cross-sectional area of the steam piping at this location is larger than the sum of the flow restrictors in the intact-loop steam generators. Therefore, the larger cross-sectional area of the ruptured steamline expels steam faster than the smaller cross-sectional area of the intact-loop steam generator flow restrictors can fill it. Thus, the blowdown of the initial steam in the steamline header piping is modeled in the first few seconds of the event, followed by the reverse-flow blowdown from the intact-loop steam generators until MSIV closure. The initial reverse-flow blowdown is discussed in subsection 6.5.1.2.8, Steamline Volume Blowdown, and provided in Table 6.5.1-4.

The full DEGB represents the break producing the highest mass flowrate from the faulted-loop steam generator. Smaller DEGB break sizes are represented by a reduction in the initial steam blowdown rate at the time of the break. Therefore, no other DEGB break sizes have been considered other than the full DEGB.

For the split-break MSLB cases, the break area is smaller than the area of a single integral flow restrictor. The flowrate from all steam generators prior to MSIV closure and the flowrate from a single steam generator after MSIV closure supply the steam flow to the break. The steam in the unisolable portion of the steamline does not affect the blowdown until the time of steam generator dry out, when the flowrate from the steam generator would decrease below the critical flowrate out of the break. At this point, the additional steam in the piping begins to have an effect on break flowrate until the steamline piping is empty. To model this effect in LOFTRAN, the mass of the unisolable steam in the steamline is added to the initial mass of the faulted steam generator. This accurately reflects both the total mass and energy that will be released from the break, and the timing of the effect of the unisolable steamline volume on the blowdown.

All the cross-sectional split-rupture areas have been redefined based on the assumption of operation with the Westinghouse-design Model Delta 75 replacement steam generators. Each break size as a function of power is the largest area that does not produce a steamline isolation signal from the Westinghouse SSPS, nor result in water entrainment in the break effluent as discussed in Reference 1.

6.5.1.3 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a mass and energy release inside containment is based on an analysis that provides sufficient conservatism to assure that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, main and auxiliary feedwater flow, steamline and feedwater isolation, and single failure such that the containment peak pressure and temperature are maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in Reference 1 and subsection 6.5.1.2 of this report. The tables of mass and energy release data for each of the steamline break cases noted in the previous section are used as input to a containment response calculation to confirm the design parameters of the HNP containment structure.

6.5.1.4 Results

Using Reference 1 as a basis, including parameter changes associated with the Model Delta 75 replacement steam generators and power uprating, the mass and energy release rates for each of the steamline break cases noted in subsection 6.5.1.2.16 have been developed for use in containment pressure and temperature response analyses. For cases that credit the High-1 and High-2 containment pressure signals, the actuation times were provided informally based on preliminary (shortened) transients. The final split-rupture cases reflect actuation times that are conservative with respect to the preliminary actuation times.

6.5.1.5 Conclusions

The mass and energy releases from the 12 steamline break cases have been analyzed at the conditions defined by the steam generator replacement and uprated power level. The analysis methods delineated in subsection 6.5.1.2 have been included in the steamline break analysis such that the results are consistent with and continue to comply with the current HNP licensing-basis/acceptance requirements. The 102-percent-power and part-power MSLB M&E releases inside containment obtained with the Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt were generated using a method consistent with MSLB

M&E releases that would be obtained with the Model Delta 75 replacement steam generators at the current NSSS power of 2787.4 MWt. The steam mass and energy releases discussed in this section have been provided for use in the containment response analysis in support of the HNP SGR/Uprating program. The results of the containment response analysis are provided in the BOP Licensing Report.

6.5.1.6 References

1. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

2. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

3. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), "LOFTRAN Code Description," April 1984.

Table 6.5.1-1 Nominal Plant Parameters for SGR/Uprating* (MSLB M&E Releases Inside Containment)	
Nominal Conditions	
NSSS Power, MWt	2912.4
Core Power, MWt	2900
Reactor Coolant Pump Heat, MWt	12.4
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	277,800
Pressurizer Pressure, psia	2250
Core Bypass, %	7.1
Reactor Coolant Vessel Average Temperature, °F	588.8*
Steam Generator**	
Steam Temperature, °F	545.9
Steam Pressure, psia	1011
Steam Flow, 10^6 lbm/hr (Plant Total)	12.87
Feedwater Temperature, °F	440
Zero-Load Temperature, °F	557

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.

** Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure. This data also corresponds to best-estimate RCS flow conditions.

Table 6.5.1-2 Initial Condition Assumptions for SGR/Uprating* (MSLB M&E Releases Inside Containment)				
Initial Conditions	**Power Level (%)**			
Parameter	**102**	**70**	**30**	**0**
RCS Average Temperature (°F)	594.8*	585.3*	572.5*	557.0
RCS Flowrate (gpm) (Thermal Design Flow)	277,800	277,800	277,800	277,800
RCS Pressure (psia)	2250	2250	2250	2250
Pressurizer Water Volume (ft^3)	836.5	704.9	529.4	397.8
Feedwater Enthalpy (Btu/lbm)	421.7	371.7	302.3	90.6
SG Pressure (psia)**	1059	1082	1115	1093
SG Water Level, faulted/intact (% NRS)	66.3/48.7	66.3/48.7	66.3/48.7	66.3/48.7

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window; temperatures include applicable calorimetric uncertainties.

** The noted SG pressures are determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties except at 0% power.

Table 6.5.1-3 Protection System Actuation Signals and Safety System Setpoints for SGR/Uprating (MSLB M&E Releases Inside Containment)	
<u>Reactor Trip</u> 2/3 Low Pressurizer Pressure – 1935 psia Safety Injection	
<u>Safety Injection</u> 2/3 Low Pressurizer Pressure – 1714.3 psia 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 High Containment Pressure - *	(implicit - used in the containment response analysis)
<u>Steamline Isolation</u> 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 High-High Containment Pressure - *	(implicit - used in the containment response analysis)
<u>Feedwater Isolation and Auxiliary Feedwater Initiation</u> Safety Injection	

* Setpoint is not explicitly modeled in MSLB M&E release analyses.

Table 6.5.1-4

Mass and Energy Flowrates for Steam Piping Reverse Flow Blowdown – Applicable to the DEGB MSLBs Inside Containment

Power Level	102%	70%	30%	0%
Steam Mass Flowrate (lbm/sec)	10,743.8	11,022.3	11,398.9	11,162.0
Steam Energy Flowrate (10^6 Btu/sec)	12.795	13.115	13.547	13.275
Duration of Blowdown (sec)	2.084	2.088	2.094	2.090

6.5.2 Main Steamline Break Mass and Energy Releases Outside Containment

6.5.2.1 Introduction

Steamline ruptures occurring outside the reactor containment structure may result in significant releases of high-energy fluid to the structures surrounding the steam systems. Superheated steam blowdowns following the steamline break have the potential to raise compartment temperatures outside containment. Early uncovery of the steam generator tube bundle maximizes the enthalpy of the superheated steam releases out the break. The impact of the steam releases depends on the plant configuration at the time of the break, the plant response to the break, as well as the size and location of the break. Because of the interrelationship among many of the factors that influence steamline break mass and energy releases, an appropriate determination of a single limiting case with respect to mass and energy releases cannot be made. Therefore, it is necessary to analyze the steamline break event outside containment for a range of conditions.

6.5.2.2 Description of Analyses

The description of the analysis and methods pertaining to the main steamline break mass and energy releases outside containment are presented below.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectra of both variables have been evaluated as part of the methodology development program documented in Reference 1. At plant power levels of 102 percent and 70 percent, various break sizes have been defined ranging from 0.1 ft^2 to the full double-ended rupture of a main steamline.

A full break spectrum at both power levels (102 percent and 70 percent) has been analyzed at the conditions associated with the Westinghouse-design Model Delta 75 steam generator replacement and uprated power. Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

6.5.2.2.1 Initial Power Level

The initial power that is assumed for steamline break analyses outside containment affects the mass and energy releases and steam generator tube bundle uncovery in two ways. First, the steam generator mass inventory increases with decreasing power levels; this will tend to delay uncovery of the steam generator tube bundle, although the increased steam pressure associated with lower power levels will cause a faster blowdown at the beginning of the transient. Second, the amount of stored energy and decay heat, as well as feedwater temperature, are less for lower power levels; this will result in lower primary temperatures and less primary-to-secondary heat transfer during the steamline break event.

The following power levels were used in the analysis:

- Full power - maximum allowable NSSS power plus uncertainty, i.e., 102 percent of rated power; and

- Near full-power - 70 percent of maximum allowable NSSS power.

For this steam generator replacement and power uprate analysis, the power levels and steamline break sizes are noted in subsection 6.5.2.2.16 of this report.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.5.2-1 and 6.5.2-2 identify the values assumed for RCS pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, and feedwater enthalpy corresponding to each power level analyzed. Steamline break mass releases and superheated steam enthalpies assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, there is a larger value for the superheated steam enthalpy available for release outside containment. The thermal design flowrate has been used for the RCS flow input consistent with the assumptions documented in Reference 2. The thermal design flowrate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the SGR/Uprating program have been applied only to the RCS average temperature (6°F), the steam generator mass (8 percent narrow-range span), and the power fraction (2 percent) and feedwater enthalpy (2°F) at full power. Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the enthalpy of superheated steam discharged out of the break.

6.5.2.2.2 Single-Failure Assumption

The limiting single failure is the failure of one train of safety functions resulting in minimum auxiliary feedwater (AFW) flow and minimum safety injection capability. Variations in AFW flow can affect steamline break mass and energy releases in a number of ways including break mass flowrate, RCS temperature, tube bundle uncovery time and steam superheating. The failure in the AFW system results in a minimum AFW flow to the steam generators; the minimum AFW flow used in the analysis is conservatively based on only one motor-driven AFW pump. The safety injection assumptions are presented in Section 6.5.2.2.11.

6.5.2.2.3 Main Feedwater System

The rapid depressurization that typically occurs following a steamline rupture results in large amounts of water being added to the steam generators through the main feedwater system. However, main feedwater flow has been conservatively modeled by assuming no increase in feedwater flow in response to the increases in steam flow following the steamline break event. This minimizes the total mass addition and associated cooling effects in the steam generators and causes the earliest onset of superheated steam released out of the break.

Isolation of the main feedwater flow is conservatively assumed to be coincident with reactor trip, irrespective of the function that produced the reactor trip signal. This assumption reduces the total mass addition to the steam generators. Closing of the feedwater flow control valves in the main feedwater lines is assumed to be instantaneous with no consideration of associated signal processing or valve stroke time.

Steamline break mass and energy releases assuming a main feedwater temperature at the high end of the feedwater temperature window are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release outside containment.

6.5.2.2.4 Auxiliary Feedwater System

Generally, within the first few minutes following a steamline break the AFW system (i.e., motor-driven AFW pumps) is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available to cover the tube bundle and reduces the amount of superheated steam produced. For this reason, AFW flow is minimized while actuation delays are maximized to accentuate the depletion of the initial secondary-side inventory. The volume of the AFW piping is maximized. A purging of the AFW piping is assumed, since maximum volume delays the injection of colder AFW into the steam generator, following any hotter feedwater resident in the piping up to the isolation valve closest to the steam generator. The less dense resident auxiliary feedwater exhibits a decreased mass addition to the faulted-loop steam generator than if the AFW is introduced directly into the steam generator. The large volume also delays the introduction of colder AFW into any steam generator, which reduces the amount of the cooldown effect on the primary side of the RCS.

6.5.2.2.5 Steam Generator Fluid Mass

A minimum initial steam generator mass in all the steam generators has been used in all of the analyzed cases. The use of a reduced initial steam generator mass minimizes the availability of the heat sink afforded by the steam generators and leads to earlier tube bundle uncovery. The initial mass has been calculated as the value corresponding to the programmed –8 percent narrow-range span level. All steam generator fluid masses are calculated assuming 0 percent tube plugging. This assumption is conservative with respect to the RCS cooldown through the steam generators resulting from the steamline break.

6.5.2.2.6 Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops become sources of energy that can be transferred to the steam generator with the broken steamline. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steamline. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.5.2.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an f(l / D) = 0 is used].

6.5.2.2.8 Steamline Volume Blowdown

There is no contribution to the mass and energy releases from the steam in the secondary plant main steam loop piping and header because the initial volume is saturated steam. With the focus of the MSLB analysis outside containment on maximizing the superheated steam enthalpy, it is presumed that the saturated steam in the loop piping and the header has no adverse effects on the results. The blowdown of the steam in this volume serves to delay the time of tube uncovery in the steam generators and is conservatively ignored. Additional information on the effect of main steam isolation on the superheated steam blowdown is discussed in subsection 6.5.2.2.9.

6.5.2.2.9 Main Steamline Isolation

Steamline isolation is assumed to terminate the blowdown from the intact-loop steam generators; the faulted-loop steam generator is assumed to be unisolable, an indication that the steamline break is upstream of the MSIV. If the MSLB is postulated downstream of the MSIV, the analysis results conservatively account for the continued blowdown from the faulted-loop steam generator as well as minimum AFW flow as discussed in subsection 6.5.2.2.4. However, there is no single failure that would permit more than one steam generator from blowing down through the pipe break.

The crediting of MSIV closure to trap superheated steam is not relevant to this analysis. MSIV closure functions only to isolate the other steam generators from the break location.

The main steamline isolation function is accomplished via the main steam isolation valve in each of the three steamlines. The actuation signal to isolate the main steamlines is received from a dynamically compensated low steamline pressure setpoint. A delay time of 7 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed.

6.5.2.2.10 Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident outside containment include reactor trip, safety injection, steamline isolation, and auxiliary feedwater. The protection system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 6.5.2-3. The setpoints used are conservative values with respect to the plant-specific values delineated in the HNP Technical Specifications.

At 102 percent power for break sizes 0.7 ft^2 and larger, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in the faulted loop. The Low Steamline Pressure signal initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip; motor-driven AFW initiation occurs as a result of the safety injection signal. However, isolation of the AFW flow to the faulted-loop steam generator occurs prior to initiation of the AFW pumps. Isolation of AFW occurs following a steamline ΔP signal. For intermediate-size breaks, from 0.6 ft^2 to 0.4 ft^2, reactor trip is actuated following the Overpower ΔT signal; for break sizes smaller than 0.4 ft^2, reactor trip is actuated following the Low-Low Steam Generator Water Level signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip. Safety injection is started as a result of a Low Pressurizer Pressure signal; steamline isolation occurs later due to Low Steamline Pressure. For the smallest break size analyzed, 0.1 ft^2, the Low Pressurizer Pressure setpoint is not reached; safety injection and steamline isolation occur as a result of Low Steamline Pressure. Auxiliary feedwater flow is initiated following the Low-Low Steam Generator Water Level signal.

At 70 percent power for break sizes 0.8 ft^2 and larger, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in the faulted loop. The Low Steamline Pressure signal initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip; motor-driven AFW initiation occurs as a result of the safety injection signal. However, isolation of the AFW flow to the faulted-loop steam generator occurs prior to initiation of the AFW pumps. Isolation of AFW occurs following a steamline ΔP signal. For break sizes smaller than 0.8 ft^2, reactor trip and motor-driven AFW initiation are actuated following a Low-Low Steam Generator Water Level signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip. Safety injection is started as a result of a Low Pressurizer Pressure signal; steamline isolation occurs later due to Low Steamline Pressure.

The turbine stop valve is assumed to close instantly following the reactor trip signal.

6.5.2.2.11 Safety Injection System

Minimum SIS flowrates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steamline break outside containment since the mass and energy releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.5.2.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat, however, does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.5.2.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 3). This version of the decay heat input has been applied previously to other HNP licensing-basis safety analyses (FSAR Chapter 15, Reference 15.0.10-3).

6.5.2.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steamline break analyses.

6.5.2.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions are used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.5.2.2.16 Description of Analysis Cases

The system transient that provides the break flows and enthalpies of the steam release through the steamline break outside containment has been analyzed with the LOFTRAN (Reference 4) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick-metal heat storage, and reverse steam generator heat transfer. The use of the LOFTRAN code for the analysis of the MSLB with superheated steam M&E releases is documented in Supplement 1 of WCAP-8822 (Reference 2), which has been reviewed and approved by the NRC for use in analyzing main steamline breaks, and in Reference 1 for MSLBs outside containment. The LOFTRAN code has been utilized previously for the HNP licensing-basis safety analyses.

The HNP NSSS has been analyzed to determine the transient mass releases and associated superheated steam enthalpy values outside containment following a steamline break event. The tables of mass flowrates and steam enthalpies are used as input conditions to the environmental evaluation of safety-related electrical equipment in the main steam tunnel.

The following licensing-basis cases of the MSLB outside containment have been analyzed at the noted conditions for the SGR/Uprating program.

- At 102 percent power, break sizes of 4.2, 2.0, 1.4, 1.2, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

- At 70 percent power, break sizes of 4.2, 2.0, 1.4, 1.2, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

Each MSLB outside containment is represented as a nonmechanistic split rupture (crack area). The largest break is postulated as a crack area equivalent to a single-ended pipe rupture. The actual cross-sectional flow area of the steamline outside containment is 4.94 ft^2, and the flow area of the steam header piping is 10.66 ft^2. However, since the break flowrate is limited by the total cross-sectional flow area of the three integral flow restrictors, the maximum break size is limited to 4.2 ft^2 rather than the actual pipe break area. Prior to steamline isolation, the break area is represented by the spectrum noted above. After steamline isolation, the break area is limited by the smaller of the integral steam generator flow restrictor (1.4 ft^2) or the defined break size.

6.5.2.3 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a mass and energy release outside containment is based on an analysis which provides sufficient conservatism to ensure that the equipment qualification temperature envelope is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, steamline and feedwater isolation, and main and auxiliary feedwater flow such that superheated steam resulting from tube bundle uncovery in the steam generators is accounted for and maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in subsection 6.5.2.2 of this report. The tables of mass flowrates and steam enthalpy values for each of the steamline break cases noted in the previous section are used as input to the environmental evaluation of safety-related electrical equipment in the main steam tunnel.

6.5.2.4 Results

Using the MSLB analysis methodology documented in Reference 1 as a basis, including parameter changes associated with the SGR/Uprating, the mass and energy release rates for each of the steamline break cases noted in subsection 6.5.2.2.16 have been developed for use in the environmental evaluation of safety-related electrical equipment in the main steam tunnel. Tables 6.5.2-4 and 6.5.2-5 provide the sequence of events for the various steamline break sizes at 102 percent and 70 percent power, respectively.

6.5.2.5 Conclusions

The mass releases and associated steam enthalpy values from the spectrum of steamline break cases outside containment have been analyzed at the conditions defined by the SGR/Uprating program. The analysis methods delineated in subsection 6.5.2.2 have been included in the steamline break analysis such that conservative mass and energy releases are calculated. The 102 percent power and 70 percent power MSLB M&E releases outside containment obtained with the Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt were generated using a method consistent with MSLB M&E releases that would be obtained with the Model Delta 75 replacement steam generators at the current NSSS power of 2787.4 MWt.

These main steamline break mass and energy releases have been calculated on a HNP plant-specific basis to address the superheated steam issue identified in the NRC Information Notice 84-90 (Main Steam Line Break Effect on Environmental Qualification of Equipment, December 7, 1984). The mass releases and associated steam enthalpy values discussed in this section have been provided for use in the environmental evaluation of safety-related electrical equipment in the main steam tunnel in support of the HNP SGR/Uprating program. The results of the steam tunnel analysis and environmental evaluation for the SGR/Uprating are provided in the BOP Licensing Report.

6.5.2.6 References

1. WCAP-10961, Rev. 1, (Proprietary), "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup," October 1985.

2. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

3. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

4. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), "LOFTRAN Code Description," April 1984.

Table 6.5.2-1 Nominal Plant Parameters for SGR/Uprating* (MSLB M&E Releases Outside Containment)	
Nominal Conditions	
NSSS Power, MWt	2912.4
Core Power, MWt	2900
Reactor Coolant Pump Heat, MWt	12.4
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	277,800
Pressurizer Pressure, psia	2250
Core Bypass, %	7.1
Reactor Coolant Vessel Average Temperature, °F	588.8*
Steam Generator**	
Steam Temperature, °F	545.9
Steam Pressure, psia	1011
Steam Flow, 10^6 lbm/hr (Plant Total)	12.87
Feedwater Temperature, °F	440
Zero-Load Temperature, °F	557

Notes:

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.

** Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure. This data also corresponds to best-estimate RCS flow conditions.

Table 6.5.2-2 Initial Condition Assumptions for SGR/Uprating* (MSLB M&E Releases Outside Containment)		
Initial Conditions	**Power Level (%)**	
Parameter	**102**	**70**
RCS Average Temperature (°F)	594.8*	585.3*
RCS Flowrate (gpm) (Thermal Design Flow)	277,800	277,800
RCS Pressure (psia)	2250	2250
Pressurizer Water Volume (ft^3)	836.5	704.9
Feedwater Enthalpy (Btu/lbm)	421.7	371.7
SG Pressure (psia)**	1059	1082
SG Water Level, faulted/intact (% NRS)	48.7	48.7

Notes:

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window; temperatures include applicable calorimetric uncertainties.

** The noted SG pressures are determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties except at 0% power.

Table 6.5.2-3 **Protection System Actuation Signals and Safety System Setpoints for SGR/Uprating (MSLB M&E Releases Outside Containment)**
Reactor Trip 2/3 Low-Low Steam Generator Water Level in any loop – 0% narrow-range span 2/3 Low Pressurizer Pressure - 1935 psia 2/4 Power-Range High Neutron Flux – 118% rated thermal power 2/3 Overtemperature ΔT K1 = 1.32 K2 = 0.0224 K3 = 0.0012 dynamic compensation lead - 22 seconds lag - 4 seconds 2/3 Overpower ΔT K4 = 1.18 K5 = 0.0 K6 = 0.0 dynamic compensation rate lag - 13 seconds Safety Injection
Safety Injection 2/3 Low Pressurizer Pressure – 1714.3 psia 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds
Steamline Isolation 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds
Feedwater Isolation Coincident with Reactor Trip (conservative assumption)
Auxiliary Feedwater Initiation (motor-driven AFW pumps) 2/3 Low-Low Steam Generator Water Level in any loop – 0% narrow-range span Safety Injection

Table 6.5.2-4

Transient Summary for the Spectrum of Breaks at 102% Power - Outside Containment

Power Level (% Nom)	Break Size (ft^2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Steamline Isolation (sec)	Time of AFW Actuation (sec)	Time of AFW Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
102	4.2	LSP	2.9	2.9	LSP	27.9	7.9	62.4	8.5	45.5
102	2.0	LSP	3.2	3.2	LSP	28.2	8.2	62.7	8.5	47.0
102	1.4	LSP	3.8	3.8	LSP	28.8	8.8	63.3	9.2	48.5
102	1.2	LSP	4.3	4.3	LSP	29.3	9.3	63.8	9.7	53.0
102	1.0	LSP	5.1	5.1	LSP	30.1	10.1	64.6	10.5	59.0
102	0.9	LSP	5.7	5.7	LSP	30.7	10.7	65.2	11.2	63.0
102	0.8	LSP	6.8	6.8	LSP	31.8	11.8	66.3	12.3	68.5
102	0.7	LSP	10.3	10.3	LSP	35.3	15.3	69.8	15.5	76.0
102	0.6	OPΔT	28.0	28.0	LPP	107.9	151.8	115.8	153.0	167.5
102	0.5	OPΔT	33.6	33.6	LPP	128.0	272.8	125.5	273.9	185.0
102	0.4	OPΔT	48.6	48.6	LPP	165.6	338.9	141.4	340.1	211.5
102	0.3	LSGWL	119.9	119.9	LPP	274.6	404.8	177.9	406.4	226.3
102	0.2	LSGWL	173.8	173.8	LPP	419.3	599.2	231.8	603.3	304.0
102	0.1	LSGWL	337.4	337.4	LSP	1136.6	1116.6	395.4	1122.8	520.5

LSP - Low Steamline Pressure

LSGWL - Low-Low Steam Generator Water Level

LPP - Low Pressurizer Pressure

OPΔT - Overpower ΔT

Table 6.5.2-5 Harris Nuclear Plant Transient Summary for the Spectrum of Breaks at 70% Power - Outside Containment										
Power Level (% Nom)	**Break Size (ft^2)**	**Reactor Trip Signal**	**Time of Rod Motion (sec)**	**Feedwater Isolation (sec)**	**Safety Injection Signal**	**Time Safety Injection Starts (sec)**	**Steamline Isolation (sec)**	**Time of AFW Actuation (sec)**	**Time of AFW Isolation (sec)**	**SG Tube Uncovery in Faulted SG (sec)**
70	4.2	LSP	2.7	2.7	LSP	27.7	7.7	62.2	8.4	57.0
70	2.0	LSP	3.3	3.3	LSP	28.3	8.3	62.8	8.7	60.0
70	1.4	LSP	4.0	4.0	LSP	29.0	9.0	63.5	9.5	61.0
70	1.2	LSP	4.5	4.5	LSP	29.5	9.5	64.0	10.0	66.5
70	1.0	LSP	5.4	5.4	LSP	30.4	10.4	64.9	10.9	74.5
70	0.9	LSP	6.1	6.1	LSP	31.1	11.1	65.6	11.7	79.5
70	0.8	LSP	7.5	7.5	LSP	32.5	12.5	67.0	13.1	87.0
70	0.7	LSGWL	54.6	54.6	LPP	124.3	117.5	112.6	118.4	152.0
70	0.6	LSGWL	62.3	62.3	LPP	141.7	157.5	120.3	157.4	188.0
70	0.5	LSGWL	72.9	72.9	LPP	166.4	317.3	130.9	316.4	212.0
70	0.4	LSGWL	88.8	88.8	LPP	203.2	398.4	146.8	397.2	254.0
70	0.3	LSGWL	114.9	114.9	LPP	265.9	520.9	172.9	523.3	314.0
70	0.2	LSGWL	166.7	166.7	LPP	399.1	758.6	224.7	762.4	380.3
70	0.1	LSGWL	320.9	320.9	LPP	890.5	1497.8	378.9	1515.0	566.8

LSP - Low Steamline Pressure
LSGWL - Low-Low Steam Generator Water Level
LPP - Low Pressurizer Pressure

6.5.3 Steam Releases for Radiological Dose Analysis

6.5.3.1 Introduction

Vented steam releases have been calculated for the Loss-of-Offsite AC Power, Locked Rotor, and Steamline Break events to support the SGR/Uprating program. Information documented in Tables 15.1.5-5 and 15.2.6-5 of the HNP FSAR includes steam releases and feedwater flows. These steam releases and feedwater flows are used as input to the radiological dose analysis that is required to support the SGR/Uprating program.

6.5.3.2 Description of Analyses

The description of the analysis and methods pertaining to the calculated steam releases and feedwater flows for use in the radiological analysis are presented below.

Steam dump will be required until the reactor can be placed on the residual heat removal (RHR) system. It has been confirmed that eight hours of steam release will occur prior to placing the plant in the RHR mode of operation. In the event of the loss of one train of electric power, the power supply to one of the two RHR loop suction valves in an RHR train must be rewired to the functioning train of electric power. This problem is assumed to be identified and remedied within the 8-hour time frame of the steam releases from the steam generator PORVs. After the first 2 hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 6 hours are required to cool down and depressurize from no-load conditions to the RHR operating conditions.

The Steam Generator Blowdown System was assumed to be isolated for the events analyzed. This is conservative with regard to the calculation of the steam released to the atmosphere.

The amount of steam released to the atmosphere depends on the sensible heat and decay heat generated while reducing the temperature from the full-power value to the shutdown conditions. A calculation is performed to determine the amount of steam that is dissipated through the atmospheric steam release.

The total RCS energy at the end of the first 2-hour interval is subtracted from the sum of the initial RCS energy and the decay heat generated during this interval. For the Steamline Break event, it is conservative to assume that the contents of the faulted-loop steam generator blow down within the first 2 hours with no energy extraction from the RCS (i.e., no temperature decrease) due to the blowdown. Likewise, the total RCS energy at the end of the 2-to-8-hour interval is subtracted from the sum of the RCS energy and the decay heat generated during this 6-hour interval.

An energy balance during both of these intervals is used to calculate the mass of auxiliary feedwater injected during the cooldown interval. The mass of feedwater injected is used to calculate the steam mass vented to the environment through the intact-loop steam generators. For the Loss-of-Offsite AC Power and Locked Rotor events, three intact loops have been used in the steam release calculations; for the Steamline Break event, two intact loops have been used for this calculation. An additional calculation is performed for the Steamline Break event in which the contents of the faulted-loop steam generator blow down during the first 2-hour interval.

6.5.3.3 Acceptance Criteria

There are no specific acceptance criteria associated with the calculation of the steam releases and feedwater flows used as input to the radiological dose analyses. Tables of steam releases and feedwater flows for each of the cooldown intervals for each of these three transients are used as input to the radiological dose analysis in support of the HNP SGR/Uprating.

6.5.3.4 Results

Table 6.5.3-1 summarizes the vented steam releases from the intact-loop steam generators as well as feedwater flows for the 0–2 hour time period and the 2–8 hour time period for the Loss-of-Offsite AC Power, Locked Rotor, and Steamline Break events. These two time periods are documented to support the SGR/Uprating program.

For the Steamline Break event, additional steam is released through the faulted-loop steam generator from the initiation of the transient up through the time at which main and auxiliary feedwater flows are assumed to be terminated. The additional steam comes from two components: the initial mass of steam in the steam generator and feedwater addition subsequent to the initiation of the Main Steamline Break. The steam mass from the faulted-loop steam generator is 162,000 lbm. This steam generator mass is not used in the energy balance which predicts the results listed in Table 6.5.3-1. The feedwater component of the steam mass may be treated as clean since it is injected into the faulted steam generator after the initiation of the steamline break and is not subjected to any primary-to-secondary coolant leakage. The feedwater mass to the faulted steam generator is also not used in the energy balance which predicts the results in Table 6.5.3-1.

6.5.3.5 Conclusions

The steam releases and feedwater flows have been calculated at the conditions defined for the SGR/Uprating for the HNP Loss-of-Offsite AC Power, Locked Rotor, and Steamline Break events. The analysis methods delineated in subsection 6.5.3.2 have been included in the steam release calculations for each transient such that the results are consistent with and continue to comply with the current HNP licensing-basis/acceptance requirements. The calculated steam releases and feedwater flows obtained with the Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Model Delta 75 replacement steam generators at the current NSSS power of 2787.4 MWt. The results of the radiological dose analysis are provided in the BOP Licensing Report for the SGR/Uprating program.

Table 6.5.3-1 Vented Steam Released and Feedwater Flows				
Event	**Vented Steam Release**		**Feedwater Flow**	
	0–2 hours	**2–8 hours**	**0–2 hours**	**2–8 hours**
Loss-of-Offsite AC Power and Locked Rotor	364,000 lbm	939,000 lbm	508,000 lbm	1,052,000 lbm
Steamline Break	386,000 lbm	892,000 lbm	482,000 lbm	967,000 lbm

6.6 LOCA Hydraulic Forces

6.6.1 Introduction

The purpose of an analysis of the loss-of-coolant accident (LOCA) hydraulic forces is to generate the hydraulic forcing functions that occur on Reactor Coolant System (RCS) components as a result of a postulated LOCA.

The hydraulic forcing functions that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The limiting break location and area vary with the RCS component under consideration, but historically the limiting postulated breaks are a limited displacement reactor pressure vessel (RPV) inlet/outlet nozzle break or a double-ended guillotine (DEG) reactor coolant pump (RCP)/steam generator (SG) inlet/outlet nozzle break (Reference 1). The NRC's recent revision to General Design Criteria (GDC)-4 allows main coolant piping breaks to be "excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." This exemption is generally referred to as leak-before-break (LBB). For HNP, the applicability of a leak-before-break design basis was approved in Reference 2. The technical justification for application of LBB to HNP is documented in Reference 3. LBB licensing allows RCS components to be evaluated for LOCA integrity considering the next most limiting auxiliary line breaks.

6.6.2 Description of Analyses and Evaluations

For the LOCA hydraulic forces analysis, the plant parameters considered to be most critical are: pressurizer pressure (2288 psia including uncertainty) and RCS hot leg (T_{hot} = 600.6°F including uncertainty) and cold leg (T_{cold} = 529.8°F including uncertainty) temperatures associated with full-power operation. Other plant parameters that are of importance for the LOCA hydraulic forces are primary side geometry and hydraulic losses in the RCS, and inputs (such as fuel mass and stiffness, and steam generator vertical divider plate mass and natural frequency in air), which affect the flexible walls modeled (core barrel and steam generator vertical divider plate).

The NRC-approved MULTIFLEX computer code (Reference 4) is used to generate the transient hydraulic forcing functions on the reactor vessel and internals due to a postulated rupture in the RCS. Hydraulic forcing functions on the RCS loop piping and steam generators are evaluated using the established LOCA forces sensitivities to changes in RCS temperatures, and the reduced break area associated with LBB licensing. The MULTIFLEX code calculates the thermal-hydraulic transient within the RCS and considers subcooled, transition and two-phase (saturated) blowdown regimes. The code employs the method of characteristics to solve the conservation laws, assuming one dimensional flow and a homogeneous liquid-vapor mixture. The RCS is divided into subregions in which each subregion is regarded as an equivalent pipe. A complex network of these equivalent pipes is used to represent the entire primary RCS.

A coupled fluid-structure interaction is incorporated into the MULTIFLEX code by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. For the reactor vessel/internals analysis, the reactor core barrel is modeled as an equivalent beam with the structural properties of the core barrel in a plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into ten segments, with each segment consisting of three walls. Mass and stiffness matrices that are obtained from an independent modal analysis of the reactor core barrel are applied in the equations of structural vibration at each of the ten mass point locations. Horizontal forces are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative of the ten mass points of the beam model. The resultant core barrel motion is then translated into an equivalent change in flow area in each downcomer annulus flow channel. At every time increment, the code iterates between the hydraulic and structural subroutines of the program at each location confined by a flexible wall. For the reactor pressure vessel and specific vessel internal components, the MULTIFLEX code generates the LOCA pressure transient that is input to the LATFORC and FORCE2 post-processing codes (Reference 4). These codes, in turn, are used to calculate the actual forces on the various components.

Steam generator hydraulic transient time history data is extracted directly from the MULTIFLEX output, as described in Reference 5.

The LATFORC computer code employs the field pressures generated by MULTIFLEX code, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall, core barrel, and thermal shield. The LATFORC code represents the downcomer region with a model that is consistent with the model used in the MULTIFLEX blowdown calculations. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The results of the MULTIFLEX/LATFORC analysis of the horizontal forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with vertical LOCA hydraulic forces, seismic, thermal, and system shaking loads, are used by the cognizant structural groups to determine the resultant mechanical loads on the reactor pressure vessel and vessel internals.

The FORCE2 computer code calculates the hydraulic forces that the RCS coolant exerts on the vessel internals in the vertical direction. The FORCE2 code uses a detailed geometric description of the vessel components and the transient pressures, mass velocities, and densities computed by the MULTIFLEX code. The analytical basis for the derivation of the mathematical equations employed in the FORCE2 code is the one-dimensional conservation of linear momentum. Note that the computed vertical forces do not include body forces on the vessel internals, such as deadweight or buoyancy. When the vertical forces on the reactor pressure vessel internals are calculated, pressure differential forces, flow stagnation forces, unrecoverable orifice losses, and friction losses on the individual components are considered. These force components are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component. The results of the MULTIFLEX/FORCE2 analysis of the vertical forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with horizontal LOCA hydraulic forces, seismic, thermal, and system shaking loads, were used in the structural evaluations previously presented to determine the resultant mechanical loads on the vessel and vessel internals.

The loop forces analysis was completed using the THRUST post-processing code. The THRUST code is used to generate the X, Y and Z directional component forces during a LOCA blowdown from the RCS pressure, density, and mass flux. The THRUST code (previously named STHRUST) is described and documented in Reference 6.

With the LBB licensing status of the HNP, only 3 of the original 11 postulated break locations will apply for the loop forces calculations. These are the three branch line breaks: accumulator line, pressurizer surge line, and RHR line. Again, the same three branch line breaks (branch lines nearest the reactor vessel) were postulated for the vessel forces calculation (the accumulator line break from loop 1 or 2, the RHR line break from loop 3, and the pressurizer line break from loop 2). For the steam generator forces calculation, the same branch line breaks were postulated; however, the branch lines nearest the steam generator were used, rather than the branch lines nearest the reactor vessel (the accumulator line break from loop 3, the RHR line break from loop 1, and the pressurizer line break from loop 2).

6.6.3 Acceptance Criteria

There are no specific acceptance criteria on results of LOCA forces analyses by themselves. LOCA Forces are used as input to other component qualification analyses.

6.6.4 Results

6.6.4.1 Reactor Vessel and Vessel Internals

Vessel and vessel internals LOCA hydraulic forcing functions were generated using three postulated auxiliary line breaks. An accumulator line break, a pressurizer surge line break and RHR line break were analyzed using a flexible beam core barrel MULTIFLEX model (for fluid-structure interaction). Vessel LOCA forces were analyzed for the HNP SGR/Uprating program with inputs that reflect the change in fuel product used at HNP from Westinghouse 17x17 Vantage 5 to Siemens Power Corporation 17x17 Fuel. As a result, this impacted the values of input parameters for modeling the core, relative to flow area and hydraulic losses. It also affected the beam model inputs representing the flexible core barrel. The vessel LOCA forces were reanalyzed and fuel/vessel qualification will be based on the new vessel forces.

6.6.4.2 RCS Loop Piping and Steam Generators

As with the vessel forces, the Loop LOCA Forces and the Steam Generator Forces were generated using three postulated auxiliary line breaks (Accumulator line, pressurize line and RHR line break). Loop hydraulic forces and Steam Generator forces had to be reanalyzed due to specific loop branch line locations and the changes in the RCS conditions of temperature and pressurizer pressure and uncertainties.

6.6.5 Conclusions

Vessel LOCA Forces were analyzed for the HNP SGR/Uprating program with inputs that reflect the change in fuel product used at HNP from Westinghouse 17x17Vantage 5 to Siemens Power Corporation 17x17 fuel. Loop LOCA Forces and Steam Generator Forces were reanalyzed due to specific loop branch line locations and HNP specific conditions of temperature and pressure.

In all cases, the basic magnitude of the LOCA hydraulic forces remained close to the values currently calculated with differences in specific components where input values have changed from the existing analyses (such as fuel type, branch line location and HNP replacement SG design). The results of the analysis, namely vessel, loop and steam generator forces at the SGR/Uprating conditions bound operation with the Delta 75 replacement steam generators at the current NSSS power level of 2787.4 MWt. The results were used by the cognizant component analysis groups for use in the project. The acceptability of the LOCA hydraulic forces is demonstrated in the structural analyses for the components as described in Sections 5.2, 5.5 and 5.7.

6.6.6 References

1. WCAP-8082-P-A, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," approved by R. Salvatori, January 1975.

2. NRC Docket 50-400, "Request for Exemption From a Portion of General Design Criteria 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as a Structural Design Basis for Shearon Harris Nuclear Power Plant, Unit 1," G. W. Knighton, June 5, 1985.

3. WCAP-14549, "Technical Justification for Eliminating Large Loop Pipe Rupture as the Structural Design Basis for the Shearon Harris Unit 1 Nuclear Power Plant," December 1996.

4. WCAP-8708-P-A, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," K. Takeuchi, et al., September 1977.

5. WCAP-7832-A, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions," P. De Rosa, April 1978.

6. WCAP-8252, Rev. 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," K.M. Vashi, May 1977.

6.7 Reactor Trip System/Engineered Safety Feature Actuation System Setpoints

6.7.1 Introduction

The Technical Specification Reactor Trip System (RTS)/Engineered Safety Feature Actuation System (ESFAS) setpoints have been reviewed for Harris Nuclear Plant (HNP) operation at SGR/Uprating conditions. As part of the review, Technical Specification changes are recommended consistent with the original Westinghouse setpoint methodology and with the original HNP licensing bases.

The Technical Specification Allowable Values are used to determine operability of the process instrumentation, and have been updated (where required) to reflect the calibration tolerances used in current HNP plant calibration procedures. HNP calibration procedures use two-sided calibration tolerances for calibration of the instrumentation. An Allowable Value must be linked to the Technical Specification Trip Setpoint to determine operability. The Allowable Value is generally determined by adding the channel's rack drift and its corresponding calibration tolerance to the Trip Setpoint (for a high trip setpoint), or by subtracting the rack drift and calibration tolerance from the Trip Setpoint (for a low trip setpoint).

6.7.2 Description of Analyses and Evaluations

The uncertainty analysis uses the "square root of the sum of the squares" to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, which are statistically independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, which can then be statistically combined.

6.7.3 Acceptance Criteria

The acceptance criteria for the RTS/ESFAS setpoints is: Margin ≥ 0.

Margin is defined as the difference between the Total Allowance (TA) and the Channel Statistical Allowance (CSA). Total Allowance is the difference between the limiting FSAR Chapter 15 safety analysis limit and the Technical Specification Trip Setpoint (in percent of instrument span). Channel Statistical Allowance is the statistical combination of the instrument channel uncertainty components (in percent of instrument span).

6.7.4 Results

Tables 6.7-1 and 6.7-2 list both the current and power uprate RTS/ESFAS setpoint values for each automatic function and parameter; required changes have been highlighted. Incorporating these changes (within Technical Specification Table 2.2.1 and Table 3.3-4) will ensure that HNP will operate in a manner consistent with the FSAR assumptions. The results of the RTS/ESFAS setpoint analysis are provided in References 1 and 2.

6.7.5 Conclusions

The results obtained with the Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Model Delta 75 replacement steam generators at the current NSSS power of 2787.4 MWt.

6.7.6 References

1. WCAP-15249, Rev. 0, "Westinghouse Protection System Setpoint Methodology for Harris Nuclear Plant (For Uprate to 2912.4 MWt - NSSS Power and Replacement Steam Generators)," January 2000.

2. CP&L Calculation HNP-I/INST-1010, Rev. 0, "Evaluation of Tech Spec Related Setpoints, Allowable Values, and Uncertainties associated with RTS/ESFAS Functions for Steam Generator Replacement (with Currrent 2787 MWt-NSSS Power or Uprate to 2912.4 MWt-NSSS Power".

Table 6.7-1 Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Power Range, Neutron Flux – High, Reactor Trip **Functional Unit 2**				
Low Setpoint	≤25.0% RTP	≤25.0% RTP	≤27.1% RTP	≤27.1% RTP
High Setpoint	≤109.0% RTP	≤109.0% RTP	≤111.1% RTP	≤111.1% RTP
Power Range, Neutron Flux, Positive Rate – High, Reactor Trip **Functional Unit 3**				
High Setpoint	≤5.0% RTP with time constant ≥2 seconds	≤5.0% RTP with time constant ≥2 seconds	≤6.3% RTP with time constant ≥2 seconds	≤6.3% RTP with time constant ≥2 seconds
Power Range, Neutron Flux, Negative Rate – High, Reactor Trip **Functional Unit 4**				
High Setpoint	≤5.0% RTP with time constant ≥2 seconds	≤5.0% RTP with time constant ≥2 seconds	≤6.3% RTP with time constant ≥2 seconds	≤6.3% RTP with time constant ≥2 seconds
Intermediate Range, Neutron Flux – High, Reactor Trip **Functional Unit 5**				
High Setpoint	≤25.0% RTP	≤25.0% RTP	≤30.9% RTP	≤30.9% RTP
Source Range, Neutron Flux – High, Reactor Trip **Functional Unit 6**				
High Setpoint	≤1.0 x 10^5 cps	≤1.0 x 10^5 cps	≤1.4 x 10^5 cps	≤1.4 x 10^5 cps

Table 6.7-1 (Cont.)
Summary of the Technical Specification Reactor Trip System Setpoint Changes

	Trip Setpoint		Allowable Value	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Overtemperature ΔT Reactor Trip Functional Unit 7				
K_1	≤1.17	≤1.185	≤2.1% ΔT	≤1.4% ΔT span for ΔT ≤2.0% ΔT span for T_{avg} ≤0.4% ΔT span for Pressure ≤0.7% ΔT span for ΔI
K_2	0.0224/°F	0.0224/°F	N/A	N/A
K_3	0.001072/psig	0.0012/psig	N/A	N/A
T′	Nominal T_{avg} ≤580.8°F	Reference T_{avg} ≤588.8°F	N/A	N/A
-ΔI Gain	2.36	1.75	N/A	N/A
+ΔI Gain	1.57	1.50	N/A	N/A
f(ΔI) Penalty Dead-band	-21.6%, to +12.0%	-21.6%, to +12.0%	N/A	N/A
τ_1	8 seconds	0 seconds	N/A	N/A
τ_2	3 seconds	0 seconds	N/A	N/A
τ_3	0 seconds	4 seconds	N/A	N/A
τ_4	20 seconds	22 seconds	N/A	N/A
τ_5	4 seconds	4 seconds	N/A	N/A
τ_6	0 seconds	0 seconds	N/A	N/A

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Overpower ΔT Reactor Trip Functional Unit 8				
K_4	≤1.079	≤1.12	≤2.3%ΔT	≤1.4%ΔT span for ΔT ≤0.2%ΔT span for T_{avg}
K_5	0.02/°F for increasing T, 0 for decreasing T	0.02/°F for increasing T, 0 for decreasing T	N/A	N/A
K_6	0.002/°F for T>T"; 0 for T≤T"	0.002/°F for T>T"; 0 for T≤T"	N/A	N/A
T″	Indicated T_{avg} ≤580.8°F	Reference T_{avg} ≤588.8°F	N/A	N/A
τ_1	8 seconds	0 seconds	N/A	N/A
τ_2	3 seconds	0 seconds	N/A	N/A
τ_3	0 seconds	4 seconds	N/A	N/A
τ_6	0 seconds	0 seconds	N/A	N/A
τ_7	10 seconds	13 seconds	N/A	N/A

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
Pressurizer Pressure – Low, Reactor Trip Functional Unit 9				
Low Setpoint	≥1960 psig	≥1960 psig	≥1946 psig	≥1948 psig
Pressurizer Pressure – High, Reactor Trip Functional Unit 10				
High Setpoint	≤2385 psig	≤2385 psig	≤2399 psig	≤2397 psig
Pressurizer Water Level – High, Reactor Trip Functional Unit 11				
High Setpoint	≤92% of span	≤92% of span	≤93.8% of span	≤93.5% of span
Reactor Coolant Flow – Low, Reactor Trip Functional Unit 12				
Low Setpoint	≥90.5% loop indicated flow[(a)]	≥90.5% loop indicated flow[(a)]	≥89.5% loop indicated flow[(a)]	≥89.5% loop indicated flow[(a)]
Steam Generator Water Level – Low-Low, Reactor Trip Functional Unit 13				
Low-Low Setpoint	≥38.5% of span	≥25.0% of span	≥36.5% of span	≥23.5% of span
Steam Generator Water Level – Low, Reactor Trip Functional Unit 14				
Low Setpoint	≥38.5% of span	≥25.0% of span	≥36.5% of span	≥23.5% of span
Coincident With Steam/Feedwater Flow Mismatch – High Functional Unit 14				
High Setpoint	≤40.0% of full steam flow at RTP	≤40.0% of full steam flow at RTP	≤43.1% of full steam flow at RTP	≤43.1% of full steam flow at RTP

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Undervoltage – RCPs, Reactor Trip Functional Unit 15				
Low Setpoint	≥5148 volts	≥5148 volts	≥4920 volts	≥4920 volts
Underfrequency – RCPs, Reactor Trip Functional Unit 16				
Low Setpoint	≥57.5 Hz	≥57.5 Hz	≥57.3 Hz	≥57.3 Hz
Turbine Trip - Low Fluid Oil Pressure Functional Unit 17.a				
Low Setpoint	≥1000 psig	≥1000 psig	≥950 psig	≥950 psig
Turbine Trip - Turbine Throttle Valve Closure Functional Unit 17.b				
Low Setpoint	≥1% open	≥1% open	≥1% open	≥1% open
Reactor Trip System Interlocks Functional Unit 19.a				
Intermediate Range Neutron Flux, P-6	$\geq 1\text{x}10^{-10}$ amp	$\geq 1\text{x}10^{-10}$ amp	$\geq 6\text{x}10^{-11}$ amp	$\geq 6\text{x}10^{-11}$ amp
Functional Unit 19.b.1				
Low Power Reactor Trips Block, P-7 (P-10 Input)	≤10.0% RTP	≤10.0% RTP	≤12.1% RTP	≤12.1% RTP
Functional Unit 19.b.2				
Low Power Reactor Trips Block, P-7 (P-13 Input)	≤10.0% RTP Turbine Impulse Pressure Equivalent	≤10.0% RTP Turbine Impulse Pressure Equivalent	≤12.1% RTP Turbine Impulse Pressure Equivalent	≤12.1% RTP Turbine Impulse Pressure Equivalent

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
Reactor Trip System Interlocks **Functional Unit 19.c**				
Power Range Neutron Flux, P-8	≤49.0% RTP	≤49.0% RTP	≤51.1% RTP	≤51.1% RTP
Functional Unit 19.d				
Power Range Neutron Flux, P-10	≥10.0% RTP	≥10.0% RTP	≥7.9% RTP	≥7.9% RTP
Functional Unit 19.e				
Turbine Impulse Chamber Pressure, P-13	≤10.0% RTP Turbine Impulse Pressure Equivalent	≤10.0% RTP Turbine Impulse Pressure Equivalent	≤12.1% RTP Turbine Impulse Pressure Equivalent	≤12.1% RTP Turbine Impulse Pressure Equivalent

Notes:

(a) Minimum Measured Flow is 97,847 gpm/loop

Table 6.7-2 Summary of the Technical Specification ESFAS Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
Containment Pressure – High 1, Safety Injection (and other functions) Functional Unit 1.c				
High Setpoint	≤3.0 psig	≤3.0 psig	≤3.6 psig	≤3.6 psig
Pressurizer Pressure – Low, Safety Injection (and other functions) Functional Unit 1.d				
Low Setpoint	≥1850 psig	≥1850 psig	≥1836 psig	≥1838 psig
Steam Line Pressure – Low, Safety Injection (and other functions) Functional Unit 1.e				
Low Setpoint	≥601 psig*	≥601 psig*	≥578.3 psig*	≥581.5 psig*
Containment Pressure – High 3, Containment Spray Functional Unit 2.c				
High Setpoint	≤10.0 psig	≤10.0 psig	≤11.0 psig	≤11.0 psig
Containment Pressure – High 2, Main Steam Line Isolation Functional Unit 4.c				
High Setpoint	≤3.0 psig	≤3.0 psig	≤3.6 psig	≤3.6 psig
Negative Steam Line Pressure Rate – High, Main Steam Line Isolation Functional Unit 4.e				
High Setpoint	≤100 psi	≤100 psi	≤122.8 psi	≤119.5 psi
Steam Generator Water Level – High-High (P-14), Turbine Trip & Feedwater Isolation Functional Unit 5.b				
High-High Setpoint	≤82.4% of span	≤78.0% of span	≤84.2% of span	≤79.5% of span

Table 6.7-2 (Cont.) Summary of the Technical Specification ESFAS Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
Steam Generator Water Level – Low-Low, Auxiliary Feedwater Functional Unit 6.c				
Low-Low Setpoint	≥38.5% of span	≥25.0% of span	≥36.5% of span	≥23.5% of span
Steam Line Differential Pressure – High, Auxiliary Feedwater Functional Unit 6.g				
High Setpoint	≤100 psi	≤100 psi	≤127.4 psi	≤127.4 psi
Safety Injection Switchover to Containment Sump Functional Unit 7.b				
Low Setpoint	≥23.4%	≥23.4%	≥20.4%	≥20.4%
Loss-of-Offsite Power – 6.9 kV Emergency Bus Undervoltage - Primary Functional Unit 9.a				
Low Setpoint	≥4830 volts with a ≤1.0 sec. time delay	≥4830 volts with a ≤1.0 sec. time delay	≥4692 volts with a ≤1.5 sec. time delay	≥4692 volts with a ≤1.5 sec. time delay
Loss-of-Offsite Power – 6.9kV Emergency Bus Undervoltage - Secondary Functional Unit 9.b				
Low Setpoint	≥6420 volts with a ≤16 sec. time delay (with Safety Injection)	≥6420 volts with a ≤16 sec. time delay (with Safety Injection)	≥6392 volts with a ≤18 sec. time delay (with Safety Injection)	≥6392 volts with a ≤18 sec. time delay (with Safety Injection)
Low Setpoint	≥6420 volts with a ≤54.0 sec. time delay (without Safety Injection)	≥6420 volts with a ≤54.0 time delay (without Safety Injection)	≥6392 volts with a ≤60.0 sec. time delay (without Safety Injection)	≥6392 volts with a ≤60.0 sec. time delay (without Safety Injection)

Table 6.7-2 (Cont.) Summary of the Technical Specification ESFAS Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
ESFAS Interlocks Functional Unit 10.a				
Pressurizer Pressure, P-11	≥2000 psig	≥2000 psig	≥1986 psig	≥1988 psig
Functional Unit 10.a				
Pressurizer Pressure, **Not** P-11	≤2000 psig	≤2000 psig	≤2014 psig	≤2012 psig
Functional Unit 10.b				
Low-Low T_{avg} P-12	≥553°F	≥553°F	≥549.3°F	≥549.3°F

Notes:

* Time constants utilized in the lead-lag controller for steam line pressure-low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

6.8 Anticipated Transient Without Scram

6.8.1 Introduction

The final Anticipated Transients Without a Reactor Scram (ATWS) rule (10CFR50.62) requires the incorporation of a diverse (from the reactor trip system) actuation of the auxiliary feedwater system and turbine trip for Westinghouse designed plants. The installation of the NRC approved ATWS Mitigating System Actuation Circuitry (AMSAC) satisfies this final ATWS rule. The basis for this rule and the AMSAC design are supported by Westinghouse generic analyses documented in NS-TMA-2182 (Reference 1). These analyses were performed based on guidelines published in NUREG-0460 (1978) (Reference 2).

Reference 1 also references WCAP-8330 (Reference 3) and subsequent related documents, which formed the initial Westinghouse submittal to the NRC forATWS, and which were based on the guidelines set forth inWASH-1270 (Reference 4). HNP is currently licensed on the basis of these calculations and analyses. For SGR/Uprating conditions, the current licensing basis calculations and analyses will be evaluated for continued applicability.

Reference 1 describes the methods used in the analysis and provides reference analyses for two-loop, three-loop, and four-loop plant designs with several steam generator models (including the Model D series) available in plants at that time. The reference analysis results for the four-loop plant design are more limiting than those for the three-loop plant design and demonstrated that the Westinghouse plant designs would satisfy the proposed criteria in NUREG-0460.

The failure of the reactor scram is presumed to be a common mode failure of the control rods to insert into the core. The assumption of this common mode failure is beyond the requirement to address a single failure in the typical FSAR Chapter 15 transient analyses. In addition, the methodology of Reference 1 uses control grade equipment to mitigate consequences of the event and uses nominal system performance characteristics in the evaluation of the event.

6.8.2 Description of Analyses and Evaluations

Reference ATWS Analysis Evaluation

An analysis was performed to evaluate the effect of the HNP Model Delta 75 replacement steam generators and power uprate on the reference analysis documented in support of Reference 1. The analysis revised the reference three-loop LOFTRAN model (Reference 5) to include Model Delta 75 steam generators and evaluated the Loss of Normal Feedwater and the Loss of Load ATWS events. These are the two most limiting RCS overpressure transients documented in Reference 1. In this analysis, a detailed NOTRUMP computer code (Reference 6) was used to model the Delta 75 replacement steam generator responses to transient conditions. The method utilized an iterative analysis technique between the detailed SG model and the LOFTRAN system transient calculations as applied in the analysis supporting Reference 1.

AMSAC Setpoint Evaluation

Additional evaluations were performed to determine the effects of the SGR/Uprating and the adequacy of the AMSAC setpoints. The evaluation for SGR/Uprating effects considered the higher power level, initial primary system average temperature, and RCS flow characteristics. The AMSAC setpoints considered an actuation setpoint at 20 percent of the steam generator narrow range span (5 percent of span below the steam generator low-low level reactor trip setpoint) and a timing delay of 25 seconds. In addition, an evaluation for the arming point (C-20 permissive) was performed.

For these evaluations, the LOFTRAN code was used to bound the SG characteristics that are critical to this event. This model is used for licensing basis calculations for the Loss of Normal Feedwater, Feedline Break, and Steamline Break Mass/Energy release events where uncovering of the steam generator tube bundle is expected to occur. The LOFTRAN model was tuned to provide conservative results in comparison to the Model Delta 75 SG characteristics seen in the reference three-loop ATWS transient described previously (Reference ATWS Analysis Evaluation) and was then applied in this evaluation.

Arming Point for AMSAC

The arming point of the AMSAC system was originally recommended to be 40 percent of the nominal power level of the plant. The original basis for this setpoint was to establish a power level below which bulk RCS coolant boiling would not occur for the first 10 minutes of the ATWS transient. A set of sensitivities for the initial power level of the plant were performed based the SGR/Uprating conditions to determine the initial power level that would support this bulk RCS coolant boiling criterion.

6.8.3 Acceptance Criteria

ATWS Rule: The final ATWS rule (10CFR50.62) requires the incorporation of a diverse (from the reactor trip system) actuation of the auxiliary feedwater system and turbine trip for Westinghouse designed plants. The installation of the NRC approved AMSAC design satisfies this final ATWS rule. The bases for this rule and the AMSAC design are supported by Westinghouse analyses documented in Reference 1. The peak RCS pressures reached in these HNP analyses shall be similar to, or less than, the peak pressures reached by the four-loop model peak pressures from Reference 1. The four-loop peak pressures are the most limiting RCS pressures reached and formed the basis for the final ATWS rule.

AMSAC Setpoint Evaluation: Peak pressures reached in the RCS shall remain below the ASME Service Level C Stress limit (3200 psig) proposed in NUREG-0460.

Arming point for AMSAC: The original basis for this setpoint was to establish a power level below which bulk RCS coolant boiling would not occur for the first 10 minutes of the ATWS transient.

6.8.4 Results and Conclusions

Reference ATWS Analysis Evaluation

The transient results demonstrated that the Model Delta 75 steam generator three-loop model was less limiting than both the Model 51 steam generator three-loop and four-loop models used in the analysis supporting Reference 1. Therefore, the basis for the finalATWS rule is unaffected by implementation of the Model Delta 75 steam generators. This analysis for the Model Delta 75 replacement steam generators at the uprated NSSS power level of 2912.4 MWt bounds operation of the plant with the Model Delta 75 replacement steam generators at the current NSSS power level of 2787.4 MWt.

AMSAC Setpoint Evaluation

The analysis for the SGR/Uprating conditions and the AMSAC specific setpoints demonstrated that the peak pressures reached in the RCS would remain below the ASME Service Level C Stress limit (3200 psig) proposed in NUREG-0460. This analysis for the Model Delta 75 replacement steam generators at the uprated NSSS power level of 2912.4 MWt bounds operation of the plant with the Model Delta 75 replacement steam generators at the current NSSS power level of 2787.4 MWt.

Arming Point for AMSAC

It was determined that an initial power level of 36.5 percent of the uprated power level (NSSS power of 2912.4 MWt) would not result in bulk RCS coolant boiling within 10 minutes of the initiation of the transient and thereby satisfies the original criterion established for the AMSAC arming point. The 36.5 percent of NSSS power of 2912.4 MWt is based upon a set of sensitivity studies to determine the power level at which the original criterion would be satisfied. An arming point value less than or equal to 36.5 percent of NSSS power of 2912.4 MWt would also satisfy the criterion. The peak pressures reached during these transients were significantly below the full-power analysis cases. This recommended arming point for the AMSAC compensates for the revised conditions of the plant.

6.8.5 References

1. Westinghouse Letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.

2. NRC Staff Report NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," April 1978.

3. WCAP-8330, "Westinghouse Anticipated Transient Without Trip Analysis," August 1974.

4. NRC ReportWASH-1270, "Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors," September 1973.

5. Westinghouse WCAP-7907-P-A (Proprietary), "LOFTRAN Code Description," WCAP-7907-A (Nonproprietary), April 1984.

6. Westinghouse WCAP-10079-P-A (Proprietary), "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

7.0 NUCLEAR FUEL

7.1 Core Thermal-Hydraulic Design

7.1.0 Introduction and Background

This section describes the core thermal-hydraulic analyses and design evaluations performed in support of the Harris Nuclear Plant SGR/Uprating.

7.1.1 Thermal-Hydraulic Compatibility

The core configuration evaluated for the SGR/Uprating consists of all SPC HTP/IFM fuel. From a thermal-hydraulic and mechanical standpoint, the fuel design for the SGR/Uprating is identical to that for previous SPC fuel designs. Thus, the fuel assemblies loaded in the core for the SGR/Uprating will be thermal-hydraulically compatible.

The thermal-hydraulic performance of the SGR/Uprating core under postulated transient and accident conditions is addressed in Sections 6.1 and 6.2.

7.1.2 MDNBR Computation

The XCOBRA-IIIC computer code (References 1 and 2) is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation.

The HTP DNB correlation (Reference 3) is used to evaluate critical heat flux in the fuel assembly. The Biasi DNB correlation (Reference 4) is used in the evaluation of Main Steamline Break cases because the hydraulic conditions are outside the range of the HTP correlation.

A DNB ratio equal to the safety limit corresponds to a 95 percent probability at a 95 percent confidence level that DNB does not occur.

7.1.3 Mixed Core DNBR Penalty

A mixed core penalty is imposed on the DNBR safety limit when the core is composed of different fuel designs. The core configuration evaluated for the SGR/Uprating consists of all SPC HTP/IFM fuel and the SGR/Uprating reload fuel will be identical to previous SPC reloads. Therefore, the application of a mixed core penalty on the MDNBR limit is not required.

7.1.4 Rod Bow Impact on DNB and LHGR

The impact of rod bowing on the MDNBR and peak LHGR was evaluated using the SPC Rod Bowing Methodology (Reference 5). The objective of the analysis was to determine the threshold burnup level at which a rod bow penalty must be applied to either the MDNBR or peak

LHGR results. Based on the maximum predicted EOC assembly exposure for first cycle fuel, the limiting margins to DNBR and LHGR limits are not impacted by rod bow. Based on predicted $F_{\Delta H}$ and F_Q values, along with the corresponding assembly exposures, the limiting margins to DNBR and LHGR limits are not impacted by rod bow for previously exposed fuel. Therefore, there is no penalty on MDNBR or LHGR for SGR/Uprating.

7.1.5 DNB Propagation

According to References 6 and 7, propagation of DNB failures needs to be considered. The effect of DNB propagation has been conservatively considered in the fuel failure calculations for Condition III and IV events.

7.1.6 Revised Overtemperature ΔT/Overpower ΔT (OTΔT/OPΔT) Trip Setpoints and Core Safety Limit Lines (CSLLs)

The OTΔT and OPΔT trip functions were evaluated using the statistical setpoint methodology for Westinghouse reactors described in Reference 8. The OTΔT and OPΔT setpoints listed in Table 6.2.0-3 were verified to adequately protect SAFDLs for SGR/Uprating operation. Also, a new set of core safety limit lines (CSLLs) were generated for the SGR/Uprating. The new CSLLs are shown in Figure 7.1-1. The OTΔT and OPΔT setpoints verification and the new CSLLs for the Delta 75 replacement steam generators at the uprated core power level of 2900 MWt are valid for operation with the Delta 75 steam generators at the currently licensed core power of 2775 MWt for steam generator tube plugging levels up to 3%.

7.1.7 Fuel Centerline Melt Limit

The fuel centerline melt LHGR limit was recalculated for SGR/Uprating conditions. The fuel centerline melt limit represents the maximum allowable LHGR on a UO_2 rod to preclude centerline melt on either a UO_2 rod or a gadolinia rod.

7.1.8 Core Bypass and Guide Tube Heating

The impact on the core bypass flows and guide tube heating were evaluated for a core composed entirely of SPC HTP fuel at SGR/Uprating conditions. The results showed a slight decrease in pressure drop as compared to the existing pressure drop calculation for an all-SPC HTP core at current operating conditions. However, the impact on the bypass fractions is negligible and the bypass fractions identified in the current licensing basis still apply for SGR/Uprating conditions.

The SGR/Uprating operating conditions were determined to have negligible impact on the margin to boiling in the guide tubes. Therefore, consistent with the current design analyses, boiling is not expected to occur in the guide tubes.

7.1.9 Conclusions

Core thermal-hydraulic analyses and design evaluations were performed in support of operation at SGR/Uprating conditions. The results are consistent with the current licensing basis, with the exception of new Core Safety Limit Lines which must be incorporated into the Technical Specifications for the SGR/Uprating.

7.1.10 References

1. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983.

2. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986.

3. EMF-92-153(P)(A) and Supplement 1, HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel, Siemens Power Corporation, March 1994.

4. L. Biasi et al, "Studies on Burnout, Part 3 – A New Correlation for Round Ducts and Uniform Heating and Its Comparison with World Data," *Energia Nucleare*, Volume 14, Number 9, September 1967.

5. XN-75-32(P)(A) Supplements 1, 2, 3, and 4, Computational Procedure For Evaluating Fuel Rod Bowing, Exxon Nuclear Company, October 1983.

6. XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, Qualification of Exxon Nuclear Fuel For Extended Burnup, Exxon Nuclear Company, October 1986.

7. ANF-88-133(P)(A) and Supplement 1, Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU, Advanced Nuclear Fuels Corporation, December 1991.

8. EMF-92-081(P)(A) and Supplement 1, Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors, Siemens Power Corporation, February 1994.

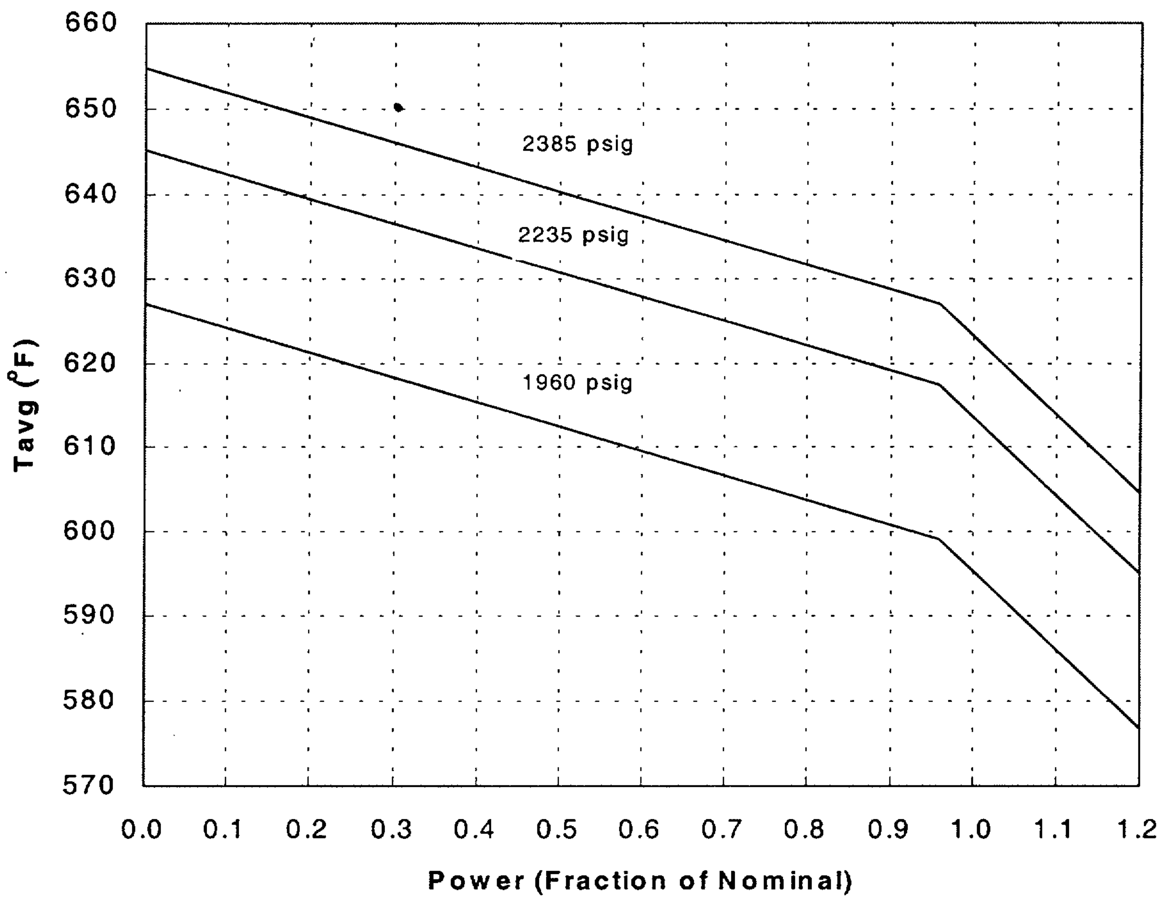

Figure 7.1-1 Core Safety Limit Lines for SGR/Uprating

7.2 Fuel Core Design

7.2.1 Introduction and Background

Siemens Power Corporation (SPC) performed the nuclear fuel and core design analyses for the Harris Steam Generator Replacement/Power Uprating (SGR/PUR) project, with input from Carolina Power & Light (CP&L). CP&L established an equilibrium fuel cycle loading pattern and fuel cycle energy operating windows, and specified this information to SPC. Based on this, SPC determined the necessary Neutronics Input to Safety in accordance with their approved methodology (Reference 7.2.6.1), and controlled this data transfer internally between their neutronics and Chapter 15 safety analysis organizations. While no new fuel assembly mechanical design changes were introduced, the enrichment, number of reload assemblies, placement of fuel assemblies in the core, peaking factor limits, and RCS operating temperature ranges were changed (Reference 7.2.6.2) relative to recent pre-uprate fuel cycles.

7.2.2 Input Parameters and Assumptions

The key nuclear analysis input parameters were transmitted from CP&L to SPC in Reference 7.2.6.2, and are summarized here in Table 7.2-1. Additional detailed thermal hydraulic inputs were provided from the Westinghouse PCWG cases (Reference 7.2.6.3), and the balance of FSAR Chapter 15 transient analysis fuel assumption information was provided to SPC in the CP&L generated Uprate Fuel Analysis Plant Parameters Document (UFAPPD), Reference 7.2.6.4. As necessary, these fuel and core related design parameters have also been transmitted in a controlled manner to other members of the Harris SGR/PUR project team, including Westinghouse.

7.2.3 Description of the Analyses

SPC performed the SGR/PUR neutronics analyses in a manner consistent with their normal reload design process (Reference 7.2.6.5). In general, this consists of a procedurally controlled process for evaluating the proposed loading patterns for adequate margins to limits, and to produce the necessary neutronics inputs to the Chapter 15 safety analyses.

Methodology

The methods and core models used in the Harris SGR/PUR analyses are described in Reference 7.2.6.1. These NRC licensed methods were used for the transition from Westinghouse fuel to SPC fuel in Harris Cycle 6, and have been used in Cycles 7 through 9. Recently NRC approved SPC methodology upgrades (replacement of XTG-PWR with PRISM) were not applied to these equilibrium cycle SGR/PUR analyses at CP&L's request. Cycle 10 (the currently operating fuel cycle) has been analyzed with PRISM, and the actual reload for Cycle 11 is also expected to be analyzed with PRISM. The reload campaign activities are expected to address the differences (as necessary) between these equilibrium SGR/PUR XTG based results, and the Cycle 11 actual loading pattern analyses. As with any other reload, neutronics parameters which deviate from the analysis of record assumptions are expected to trigger either core re-design or re-analysis of the affected transients.

The current SPC neutronics and safety analyses include specific evaluations of "mixed cores," that is, cores with a combination of standard fuel (no flow mixing spacers, typical of Westinghouse LOPAR fuel) and High Thermal Performance fuel (includes flow mixing spacers, typical of Westinghouse Vantage5 and Siemens HTP fuel designs). The SGR/PUR neutronics and safety analyses have been performed with the requirement that only SPC HTP fuel designs are allowed.

For the SGR/PUR equilibrium fuel and core design, essentially all the neutronics input to safety were re-calculated and used as the basis for a full replacement set of safety analyses. No significant changes to the nuclear design philosophy, methods or models were necessary due to the uprating.

Design Evaluation – Physics Characteristics and Key Safety Parameters

As previously discussed, an equilibrium fuel cycle design was developed by CP&L to represent future Harris SGR/PUR cores. Table 7.2-1 summarizes the key parameters that represent these cores, and provides Pre-Uprate parameters for comparison. The increased cycle energy, and the increase in RCS coolant T_{avg} are the two major contributors to changes between the equilibrium SGR/PUR fuel cycle and current operations. As can be seen from the table, the SGR/PUR peaking factor limits have been reduced by the same amount as the proposed power uprate, which tends to reduce the overall impact of the SGR/PUR program on these neutronics parameters. One impact of the higher cycle energy (required by the power uprate) is the more negative EOC Moderator Temperature Coefficient limit (from the current –45 pcm per °F to the new limit of –50 pcm per °F).

The key physics parameters and safety parameters for the SGR/PUR project described in Table 7.2-1 have been incorporated into the Chapter 15 safety analyses described in Section 6 of this report, and are shown by analysis to be acceptable. Where applicable, SPC has also evaluated SGR/PUR for the effects of a possible RCS T_{avg} reduction to the current T_{avg} value of 580.8°F, and have shown that the safety analyses remain applicable for these conditions as well.

Design Evaluation – Power Distributions and Peaking Factors

The Harris SGR/PUR fuel and core designs are based on increased overall core power (from 2775 MWt to 2900 MWt), which have been somewhat offset by a reduced allowable set of peaking factors (F-Q limit reduced from 2.52 at full power to 2.41; F-Delta-H limit reduced from 1.73 at full power to 1.66). To supply the additional energy to operate at this higher power for the same fixed time interval between refueling outages, the equilibrium fuel cycle design loads additional fuel assemblies of approximately the same enrichments that have been used at Harris in past cycles. These additional fresh fuel assemblies displace (generally) twice burned and reinserted fuel assemblies, and this leads to changes in the radial, bundle- by- bundle power distribution. The change in temperature control programs from the pre-uprate "Low Temperature Control Program" value of RCS T_{avg} of 580.8°F to 588.8°F leads to some change in the core average axial power shapes, as well.

These changes have been specifically factored into the new safety analyses in described in Section 6 of this report, and have been shown to be acceptable. Where applicable, SPC has also evaluated SGR/PUR for the effects of a possible RCS T-average reduction to the current T_{avg} value of 580.8°F, and have shown that the safety analyses remain applicable for these conditions as well.

7.2.4 Acceptance Criteria for Analyses

Except as noted (i.e., peaking factors, and MTC) there are no changes in the acceptance criteria applied by SPC to these neutronics and safety input parameters. While not a specific value, the analyses are applicable only to full cores of SPC HTP fuel design. Mixed cores of Westinghouse and SPC fuel designs are specifically not covered by these analyses. In accordance with the SPC approved methodology for reload design, most of these neutronics parameters do not have specific "acceptance criteria," other than that they must yield acceptable results in the safety analyses described in report Section 6.

7.2.5 Results

The SGR/PUR equilibrium fuel cycle design represented by the key neutronics parameters in Table 7.2-1 have been evaluated and found to be acceptable for core design as well as for input to FSAR Chapter 15 safety analyses described in Section 6 of this report.

7.2.6 References

1. "Exxon Nuclear, Neutronics Design Methods for Pressurized Water Reactors," XN-75-27(A), including Supplements 1 through 4, the latest dated December, 1986

2. "Harris SGR/PUR Fuel Cycle Design Information," CP&L letter from J. R. Caves to T. E. Millsaps at SPC, Serial NF-98A-0129, dated May 1, 1998.

3. "Harris SGR/PUR Fuel Design Information – Final PCWG Parameters," CP&L letter from R. E. Oliver to T.E. Millsaps at SPC, Serial NF-98A-214, dated August 4, 1998 (this transmits Westinghouse Letter CQL-98-030, Rev 1, "Final PCWG Parameters for the SGR/Uprating Analysis and Licensing Project," to SPC).

4. "Harris Nuclear Plant SGR/PUR Uprate Fuel Analysis Plant Parameters Document," attachment to CP&L Calculation HNP-F/NFSA-0034, Revision 3, dated 12/15/99

5. "Guidelines for PWR Neutronics Analysis," internal SPC document EMF-1356(P).

Table 7.2-1

Shearon Harris SGR/PUR Key Neutronics Input to Safety Parameters

Safety Parameter	Pre-Uprate Value	Post Uprate Value	Comment
Reactor Core Power (MWt)	2775	2900	4.5% Increase
Number of Fresh Fuel Assemblies	Typically, 56 or 60	64	This is not a design restraint or limit, but is provided for comparison purposes.
Evaluated Fuel Designs	Westinghouse LOPAR V5 Siemens Power HTP	Siemens Power HTP	Mixed cores are not analyzed by the SGR/PUR analyses.
Vessel T_{avg} (°F)	580.8	588.8	Pre-Uprate analyses allowed for EOC coastdown temperature reduction of 7°F Post Uprate analyses allow for temperature reductions between 588.8°F and 580.8°F
RCS Pressure (psia)	2250	2250	No change
Positive MTC Limit	≤ 5.0pcm/°F for power levels up to 70% full power, with a linear ramp down to 0 pcm/°F at full power	≤ 5.0pcm/°F for power levels up to 70% full power, with a linear ramp down to 0 pcm/°F at full power	
EOC negative MTC Limit (pcm/°F)	Less negative than -45	Less negative than -50	Required due to higher energy fuel cycles
F-Delta-H Peaking Factor Limit at full power	1.73	1.66	~4.5% decrease
F-Q Peaking Factor Limit at Full Power	2.52	2.41	~4.5% decrease
Required Shutdown Margin, Modes 1 and 2 (pcm)	1770	1770	No change

7.3 Fuel Rod Design and Performance

7.3.1 Introduction

The Siemens Power Corporation (SPC) advanced 17x17 fuel assembly design has been analyzed to support steam generator replacement/uprating project (SGR/Uprating) to an uprate of 4.5% at Harris Nuclear Plant (HNP). With the exception of the analyses referenced in Sections 7.3.2.3, 7.3.2.4, 7.3.2.6, 7.3.2.15.1, 7.3.2.15.2, and 7.3.2.15.3 (which support 4.5% power uprate), the mechanical analyses included an evaluation of the fuel rod assembly during both normal operation and anticipated operational occurrences (AOO) for a bounding 6% power uprate. These analyses were performed in accordance with the U. S. Nuclear Regulatory Commission (NRC)-approved methodology (References 1 and 2) to demonstrate compliance with the requirements of the SPC generic Pressurized Water Reactor (PWR) design criteria (Reference 3). These analyses evaluate the fuel rod mechanical criteria presented in the Reference 3 topical report with the following maximum discharge exposures:

- 59.8 GWd/MTU assembly exposure
- 62.0 GWd/MTU rod exposure

7.3.2 Description of Analyses and Evaluations

The fuel rod analyses consist of a group of analyses that use NRC-approved codes in addition to other support codes. The input to these analyses is provided in a file with design parameters; a file with reactor cycle and power ramp parameters; and files with the power histories.

7.3.2.1 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and the formation of hydride platelets. This failure mechanism is precluded by controlling the moisture hydrogen impurities in the rod during fabrication. Cleaning and drying of the cladding, and careful moisture control of the fuel pellets are applied to minimize the total hydrogen within the fuel rod assemblies.

7.3.2.2 Cladding Collapse

During initial fuel densification, the possibility exists for the formation of axial gaps within the fuel column. Consequently, a cladding creep analysis is performed to verify that the pellet-to-cladding gap does not close before completion of fuel densification. Creep calculations were performed according to the methodology described in Reference 1.

The plenum spring was designed to help preclude collapse. The spring design was analyzed to ensure that sufficient axial force is applied to the fuel stack to help prevent axial gaps during shipping and handling. The spring accommodates length variation due to initial fuel densification. Furthermore, the pitch of the spring is designed, using stiffening ring relationships, such that collapse in the upper plenum region is prevented.

7.3.2.3 Overheating of Cladding

To preclude overheating of the fuel rod cladding, an analysis of the design is performed to show that the thermal margin criterion for departure from nucleate boiling ratio (DNBR) is satisfied. This is addressed in Sections 6.1.5 and 6.2.

7.3.2.4 Overheating of Pellets

To prevent overheating of fuel pellets, an analysis of the design is performed to show that no pellet centerline melting occurs for normal operations and AOOs. This is addressed in Sections 6.1.5 and 6.2.

7.3.2.5 Stress and Strain Limits

Discussions on fuel rod stress and strain analyses are provided below.

7.3.2.5.1 Pellet-Cladding Interaction

To determine the stresses and strains during transients, ramps to maximum linear heat generation rate (LHGR) conditions were applied to the steady-state power histories.

All ramp rates were based on the limits from preconditioning and maneuvering criteria.

7.3.2.5.2 Cladding Strain

In addition to the ramping analysis described above, cladding strain was analyzed in both steady-state conditions and under simulated AOOs.

For steady-state conditions, the creep strain at any time throughout life was compared to a strain criterion.

Simulated AOOs were evaluated to determine the cladding total uniform strain that would occur if a rod was at the maximum allowable power peaking levels. A local power factor times a maximum overpower was applied periodically throughout irradiation.

7.3.2.5.3 Cladding Stresses

The steady-state cladding stress analysis was performed considering the same stress relationships

described in Reference 4. These relationships consider primary and secondary membrane and bending stresses due to hydrostatic pressure, flow-induced vibration, ovality, spacer contact, pellet-cladding interaction, thermal and mechanical bow, and thermal gradients.
The applicable stresses in each orthogonal direction were combined to calculate the maximum stress intensities which were then compared to the applicable American Society of Mechanical Engineers (ASME) Code design criteria given in Reference 3.

7.3.2.5.4 End Cap Stresses

An analysis was performed to determine the stresses in the end cap weld area due to the axial load of the pellet stack and plenum spring and pressure differential during normal operation and AOO conditions.

7.3.2.6 Cladding Rupture

NRC-approved cladding ballooning and rupture models are used by SPC in the evaluation of cladding rupture due to a loss of coolant accident. Compliance to the event criteria is part of the reload licensing and is evaluated for each plant, the results of which are presented in Sections 6.1.1 and 6.1.2.

7.3.2.7 Fuel Rod Mechanical Fracturing

The accident strength criteria for the fuel assembly structure is that it shall sustain, without impairing coolability or control rod insertability, the forces resulting from seismic and loss of coolant accident (LOCA) events. The loads arise from inertial forces caused by the motion of the upper and lower core plates, and lateral deflections and impacts transmitted to the assembly through adjacent assemblies, the core plates, and the core baffle.

The fuel rod stresses were determined from the combination of stresses due to steady-state operation, lateral spacer impact, and assembly axial impact.

7.3.2.8 Fuel Densification and Swelling

NRC-approved densification and swelling models are used in the SPC fuel performance codes. Using these codes, fuel densification and swelling are implicitly limited by the design criteria specified for fuel temperature, cladding strain, cladding collapse, and internal pressure.

7.3.2.9 Loading Limits

The capability of the fuel rods to withstand normal operation and anticipated operational events is analyzed. Acceptability of the assembly to withstand postulated accident conditions is discussed in Section 7.3.2.7. The capability of the design to withstand possible loads incurred during handling is described in Section 7.3.2.14. Steady-state fuel rod stresses are discussed in Section 7.3.2.5.

During fluctuations in power, the differential thermal expansion between the guide tubes and the hotter fuel rod cladding may produce tension (power decrease) or compression (power increase) in the fuel rod cladding. Using the limiting loads developed from guide tube evaluations, the safety margin to rod buckling is determined.

7.3.2.10 Fatigue

The stresses calculated in the ramping analysis (Section 7.3.2.5) were used to evaluate the cladding fatigue damage through the EOL due to the cyclic power variations. For conservatism, all cycles were considered to cause the extreme maximum stresses for the fatigue evaluation.

The transient stress results were evaluated to determine the fatigue usage for each cycle based on the O'Donnel and Langer (Reference 5) design curve. These results were accumulated to determine the total fatigue usage factor.

7.3.2.11 Oxidation, Hydriding, and Crud Buildup

External corrosion is calculated using a proprietary SPC model that includes an enhancement factor. As stated in Reference 3, effects of crud are modeled in the NRC-approved SPC fuel performance codes. The fuel rod corrosion analysis takes into account the methodology changes that occurred as the result of the review and approval of Reference 3.

7.3.2.12 Fuel Rod Growth

Projected fuel rod growth behavior is based on SPC measured data. The maximum predicted EOL rod growth is calculated using the maximum growth curve. Conservative temperatures, worst-case dimensional clearance, maximum rod growth, and minimum assembly growth are used in the analysis.

The fuel rod growth analysis takes into account the methodology changes that occurred as the result of the review and approval of Reference 3.

7.2.3.13 Rod Internal Pressure

Calculation of the rod internal pressure is performed with a proprietary SPC code. Power histories for the gas pressure analysis were developed in accordance with the NRC-approved methodology (Reference 1). For cases where system pressure is exceeded, the pellet-to-cladding gap does not increase during steady or increasing power conditions.

7.2.3.14 Fuel Assembly Handling

Handling requirements for the fuel assembly design include acceptable plenum spring axial restraint/compliance. The plenum spring is designed to prevent damage to the fuel column during shipping and handling and to aid in preventing axial gaps in the fuel column during early operation.

7.3.2.15 Fuel Coolability

The mechanical analyses for maintaining fuel coolability are described below.

7.3.2.15.1 Cladding Embrittlement

The requirements on cladding embrittlement are covered in Sections 6.1.1 and 6.1.2.

7.3.2.15.2 Violent Expulsion of Fuel

The evaluation of the deposition of energy during a reactivity-initiated severe accident is covered in Section 6.1.5.

7.3.2.15.3 Fuel Ballooning/Rupture

The analysis of clad swelling and burst strain is an integral part of the LOCA evaluation. See Sections 6.1.1 and 6.1.2.

7.3.2.15.4 Structural deformations

The mechanical analysis for maintaining fuel coolability and the capability to insert control rods is covered in Section 7.3.2.7.

7.3.3 Acceptance Criteria

Table 7.3.3-1 identifies the criteria for the SGR/Uprating evaluation. Table 7.3.3-1 includes references to the appropriate criteria given in Section 3 of the design criteria topical report (Reference 3).

7.3.4 Results

The far right-hand column of Table 7.3.3-1 provides the limiting results from the SGR/Uprating evaluation.

7.3.5 Conclusions

The analyses demonstrate that the mechanical criteria applicable to the 17x17 fuel rod design are satisfied for a 4.5% power uprate at HNP for normal operation and during AOOs subsequent to steam generator replacement. The same conclusion applies to just a steam generator replacement at HNP without power uprate. By demonstrating criteria compliance, the evaluation results continue to comply with the current HNP licensing basis/acceptance requirements.

7.3.6 References

1. XN-NF-82-06(P)(A) Revision 1, and Supplements 2, 4, and 5, Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, October 1986.

2. ANF-88-133(P)(A) and Supplement 1, Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU, Advanced Nuclear Fuels Corporation, December 1991.

3. EMF-92-116(P)(A) Revision 0, Generic Mechanical Design Criteria for PWR Fuel Designs, Siemens Power Corporation, February 1999.

4. EMF-93-074(P)(A) and Supplement 1, Generic Mechanical Licensing Report For Advanced 17x17 Fuel Design, Siemens Power Corporation, June 1994.

5. W. J. O'Donnel and B. F. Langer, "Fatigue Design Bases for Zircaloy Components," Nuclear Science and Engineering, Volume 20, January 1964.

Table 7.3.3-1 Mechanical Fuel Rod Design Evaluation Criteria and Results

Criteria Section[1]	Description	Criteria	Results or Disposition
3.2	Fuel Rod Criteria		
3.2.1	Internal hydriding	Hydrogen content in components controlled to a minimum level during manufacture to limit internal hydriding.	Controlled by manufacturing specifications and verified by Quality Control inspection. No change for uprated conditions.
3.2.2	Cladding collapse	Sufficient plenum spring deflection and cold radial gap to prevent axial gap formation during densification.	Radial gap criteria are met. Plenum spring design ensures positive downward force, and fuel rod internal pressure maintains radial gap throughout densification.
3.2.3	Overheating of cladding	95/95 confidence that fuel rods do not experience departure from nucleate boiling (DNB) during steady state or AOO.	Verified in Sections 6.1.5 and 6.2.
3.2.4	Overheating of fuel pellets	No centerline melting during normal operation and AOO.	Verified in Sections 6.1.5 and 6.2.
3.2.5	Stress and strain limits		
	Pellet/ cladding interaction (PCI)	Cladding strain meets Reference 3 criteria and no fuel melting.	AOO strain design criteria are met. Steady-state strain design criteria are met. (See Section 3.2.4, above in this table, for fuel temperatures.)
	Cladding stress	ASME Section III, Appendix III, Article III-2000, in combination with the specified 0.2% offset yield strength and ultimate strength of zircaloy.	Cladding steady-state stress design margin is maintained. The end cap weld stress design margin is maintained for normal operation and accident conditions.
3.2.6	Cladding rupture	Not underestimated during LOCA and used in determination of Code of Federal Regulations (CFR) 10 CFR 50.46 criteria.	Accepted model in evaluation. See Sections 6.1.1 and 6.1.2.

1 Section numbers correspond to Reference 3 sections.

Table 7.3.3-1 Mechanical Fuel Rod Design Evaluation Criteria and Results (Cont.)

Criteria Section[2]	Description	Criteria	Results or Disposition
3.2	Fuel Rod Criteria (Cont.)		
3.2.7	Fuel rod mechanical fracturing	ASME Section III, Appendix F.	Fuel rod stresses generated from the combination of steady-state operation and accident conditions meet the design criteria. Also see Section 3.4 in this table.
3.2.8	Fuel densification and swelling	Sections 3.2.2, 3.2.4, 3.2.5, and 3.3.7 in this table.	Models included in NRC-accepted fuel performance codes.
3.3	Fuel System Criteria		
3.3.1	Stress, strain and loading limits on assembly components. (In this table, see Section 3.3.9 for handling and Section 3.4 for accident conditions.)		
	Fuel rod	Fuel rod shall be capable of withstanding loads from normal operation and anticipated operational events.	The safety margin to rod buckling is maintained.
	Cladding	See Section 3.2.5 in this table.	See Section 3.2.5 in this table.
3.3.2	Fatigue	Cumulative usage factor (CUF) <0.67.	Cladding CUF is well below design criterion.
3.3.4	Oxidation, hydriding, and crud buildup	Acceptable maximum oxide thickness. Effects of oxidation and crud to be included in thermal and mechanical fuel rod analyses. Stress analysis to include metal loss due to oxidation.	Peak local oxide less than the design criterion. NRC-approved fuel rod performance code accounts for oxidation and crud buildup. Metal loss accounted for in cladding stress analysis. The fuel rod corrosion analysis takes into account the methodology changes that occurred as the result of the review and approval of Reference 3.
3.3	Fuel System Criteria (Cont.)		
3.3.6	Axial irradiation growth		

2 Section numbers correspond to Reference 3 sections.

Table 7.3.3-1 Mechanical Fuel Rod Design Evaluation
Criteria and Results (Cont.)

Criteria Section2	Description	Criteria	Results or Disposition
	Fuel rod	Clearance remains between fuel rod and UTP/LTP at EOL.	EOL clearance exists between fuel rods and UTP and LTP. The fuel rod growth analysis takes into account the methodology changes that occurred as the result of the review and approval of Reference 3.
3.3.7	Rod internal pressure	Acceptable maximum internal rod pressure. Gap does not open during steady state or increasing power.	Maximum internal rod pressure meets the design criterion. Gap criteria are satisfied.
3.3.9	Fuel assembly handling	The fuel rod plenum spring design provides loads to prevent fuel column damage during shipping and handling and aids in preventing axial gaps in the fuel column during operation.	Criteria are met. Plenum spring design is appropriate.
3.4	Fuel Coolability		
3.4.1	Cladding embrittlement	Include in LOCA analysis.	Accepted models in evaluation. See Sections 6.1.1 and 6.1.2.
3.4.2	Violent expulsion of fuel	<280 cal/g hottest axial deposition.	Verified in Section 6.1.5.
3.4.3	Fuel ballooning	Consider impact on flow blockage in LOCA analysis.	Accepted models in evaluation. See Sections 6.1.1 and 6.1.2.
3.4.4	Structural deformations	Maintain coolable geometry and ability to insert control rods. SRP 4.2, Appendix A, and ASME Section III, Appendix F.	Coolable geometry and control rod insertability are maintained.

* Section numbers correspond to Reference 3 sections.

7.4 Heat Generation Rates

7.4.1 Introduction and Background

Gamma ray heat generation rates in the lower core plate were specifically determined at the SGR/Uprating conditions; heat generation rates for other reactor internals components were obtained through a scaling process. The heat generation rate values were supplied as input for use in the reactor internals structural evaluations described in Section 5.2.

The presence of heat generated in the reactor internals components, along with a distribution of fluid temperatures, results in thermal gradients within and between components. These temperature gradients result in thermal stresses and thermal growth that must be accounted for in the design and analysis of the various reactor internals components. The primary design considerations are (1) to ensure that thermal growth is consistent with the functional requirements of components and (2) to ensure that the applicable ASME Code requirements are satisfied. In order to satisfy these requirements, the reactor internals components must be analyzed with respect to fatigue and maximum allowable stress considerations.

The reactor internals components that are subjected (either directly or indirectly) to significant heat generation effects are the upper and lower core plates, the lower core support, the core baffle plates, the former plates, the core barrel, the neutron pad, the baffle-former bolts, and the barrel-former bolts. Note, however, that the upper core plate, the lower core support, and the neutron pad experience little, if any, temperature rise over the surrounding reactor coolant due to relatively low heat generation rates (generally less than 50 Btu/hr-lbm).

WCAP-9620, Rev. 1 (Reference 1) provides heat generation rates for Westinghouse plants, including HNP, that were intended to be applicable for the life of the plants. However, the introduction of low leakage loading patterns has indicated that plant-specific analyses may be required for the lower core plate to assess the increased power in the radial center region of the core. Conversely, it should be noted that the power in the (radial) core periphery is reduced for low leakage loading patterns which provides margin to the WCAP-9620, Rev. 1 heating rates for components that are radially outward from the center of the core.

This section describes how the heat generation rates were determined for the HNP lower core plate at the SGR/Uprating conditions and how the component average heating rate values were determined for the remaining reactor internals components.

7.4.2 Reactor Internals Heat Generation Rates - Lower Core Plate

7.4.2.1 Description of the Analysis

Power distribution and Siemens Power Corporation (SPC) fuel geometry data has been acquired for the equilibrium reload fuel cycle and evaluated relative to data Westinghouse has previously incorporated into the heat generation rate methodology.

Conservative axial and radial power distributions were assumed in the long-term heat generation rate calculation. For the short-term bottom-peaked calculation, the axial power distribution was taken from the Core Radiation Source Data (CRSD) document as described in Reference 1. A conservative radial power distribution was used for the short-term heat generation calculation. HNP specific radial (assembly-by-assembly) loading patterns and short-term and long-term axial power distributions were also analyzed.

The analysis was performed through the use of the DORT (Reference 2,Version 3.1) discrete ordinates code, which has been used by Westinghouse to calculate heating rates for other projects that have been approved by the NRC. The lower core plate was analyzed in an R-Z (cylindrical) geometry calculation based on the equivalent volume cylindrical core concept. The varying amounts of structure located axially below the core were approximated as a number of regions each with the appropriate amount of stainless steel, water, and other materials uniformly homogenized throughout the region. The R-Z geometry included the lower three feet of the core and extended axially to one foot below the lower core plate and radially out to the inner radius of the reactor vessel.

Varied cases were used to determine the HNP lower core plate heat generation rates at SGR/Uprating conditions. Westinghouse Vantage 5H (V5H) fuel was used as the base case, but selected Siemens data for the fuel bottom nozzle and fuel rod end plug were evaluated in determining which calculation cases should be selected for the final (limiting) heat generation rates. These limiting cases were then used in the reactor vessel internals analysis.

7.4.2.2 Acceptance Criteria for Analyses

There are no acceptance criteria for this analysis, since the applicable design considerations for the reactor internals components are evaluated in subsequent calculations that use these heat generation rate results as inputs.

7.4.2.3 Results

The heat generation rates for the lower core plate were provided as input for the reactor vessel internals analysis described in Section 5.2.

Long-Term Case

The average long-term heat generation rate in the central portion of the core plate (outer radius - 59.8 inches) is 528.1 Btu/hr-lbm. In the outer annular portion (outer radius - 66.9 inches), the average is 62.7 Btu/hr-lbm.

Short-Term Case

With respect to the short-term axial power distributions, the WCAP-9620, Rev. 1 values are more conservative than the corresponding HNP data and will result in higher heat generation rates in the lower core plate.

With respect to the short-term radial distribution, a conservative radial power distribution was used for the short-term heat generation calculation.

The average short-term heat generation rate is 1403.3 Btu/hr-lbm in the inner portion of the core plate and 181.4 Btu/hr-lbm in the outer portion.

7.4.3 Radial Internals - Core Barrel, Baffle Plates, Neutron Pad

7.4.3.1 Description of the Analyses

Design basis heat generation rates that are applicable to the HNP radial internals are contained in Appendices H and I of Reference 1. The core power distributions upon which these calculations were based were derived from 25 independent fuel cycles in 11 three-loop reactors and represented an upper tolerance limit of beginning-of-cycle (BOC) and end-of-cycle (EOC) power in the peripheral fuel assemblies, based on a 95 percent probability with 95 percent confidence. The peripheral fuel assemblies are defined as those with one or two faces or one corner adjacent to the core baffle. Most of the 25 fuel cycles were out-in loading patterns that, when combined with the statistical processing selected, resulted in a core power distribution that was biased high on the core periphery. This high bias was desired by the reactor internals analysts to ensure conservative, but not unrealistic, results in the critical baffle-barrel region of the reactor internals.

Utility interest in reducing the rate of PWR reactor vessel embrittlement by reducing the incident fast neutron flux to the reactor vessel through fuel management and core periphery modifications has grown in recent years. In addition, the fuel cycle cost advantages of reduced core neutron leakage coupled with higher permissible core power peaking limits have resulted in fuel management strategies with significantly lower power levels in the peripheral fuel assemblies than was the case with traditional out-in fuel management.

The evaluation of heat generation rates for the radial internals used HNP-specific assembly-wise core power distributions for the power uprate and CRSD power distributions from Reference 1 (Figure A-3).

An assessment was made of the effect of the core power distributions on the heat generation rates in the core baffle plates and core barrel. The approach taken was to use scaling factors that account for the facts that (1) heat generation rates in the radial internals regions are the result of radiation leakage from the periphery of the core, and (2) to a close approximation, the heat generation rate in a given region is proportional to the power produced in adjacent fuel regions. These ratio expressions were determined based on discrete ordinates transport theory calculations using various core power distributions.

For the long-term axial power distribution, inputs from HNP were evaluated versus the Reference 1 long-term axial power distribution; the HNP data for long-term axial power distributions is less limiting.

For the short-term axial power distributions, comparisons show that the HNP data is less limiting than the Reference 1 axial power distributions in the regions of the core where power is being maximized for analysis.

7.4.3.2 Acceptance Criteria for Analyses

There are no acceptance criteria for the heat generation rate analysis, since the applicable design considerations for the reactor internals components are evaluated in subsequent calculations that use these heat generation rate results as inputs.

7.4.3.3 Results

For consideration of baffle plate heating, core barrel heating, and neutron pad heating, the radial power and axial power distributions were evaluated, and it was concluded that the Reference 1 heat generation rates apply (i.e., are more limiting).

Changing core power distributions affects reactor internals heat generation rates. The effect of implementing low leakage loading patterns has been examined relative to the reactor internals design basis, which assumed traditional out-in fuel management. In general, the change reduces the baffle-barrel heat generation rates by one-third to one-half. The Reference 1 distributions through the thickness of the baffle plates and core barrel are not expected to be significantly affected by the change from out-in to low leakage loading patterns.

The additional values are designed to augment the values reported in Reference 1, providing additional information on the heating rate effects of current, realistic power distributions. However, in order to retain the conservatism associated with the work performed and documented in Reference 1, it is necessary to continue to use the absolute magnitudes described in that document.

7.4.4 References

1. "Reactor Internals Heat Generation Rates and Neutron Fluences," WCAP-9620, Revision 1, A. H. Fero, December 1983.

2. RSICC Computer Code Collection CCC-650, "DOORS3.1 One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

7.5 Neutron Fluence

7.5.1 Introduction

Neutron fluence was evaluated to obtain a realistic estimate of the increase in the capsule fluence and the vessel fluence that would result from implementation of the Harris Nuclear Plant (HNP) Steam Generator Replacement and Power Uprate Project (SGR/Uprate).

7.5.2 Description of Analyses and Evaluations

To increase reactor core power, it is necessary to place once-burned or fresh unburned fuel assemblies on or close to the core periphery. This generally results in an increase in the relative power in the peripheral assemblies, and therefore an increase in the fast leakage flux. The SGR/Uprate configuration will also result in changes to the thermal-hydraulic parameters of the primary coolant system, which causes an increase in water temperatures, and put fresh (unburned fuel assemblies – one side) close to, or on the core periphery. Higher water temperatures result in a lower neutron thermalization rate, and thus an increase in the fast (E>1.0 MeV) leakage flux. These increases in fast flux result in increases in the fast fluence on the reactor vessel and on the surveillance capsule (test) specimens. These impacts are evaluated in detail in Reference 3, which is an attachment to this licensing report section. (Reference 3 is a supplementary report to the reactor vessel material surveillance report, BAW-2355, October 1999, which was submitted to the NRC via CP&L Letter, HNP-99-157 dated November 9, 1999).

In performing the fast neutron exposure evaluations for the surveillance capsules and reactor vessel, explicit values of the fast flux and fluence (E>1.0MeV) were computed for various vessel wall depths at the following locations:

- Vessel inside surface maximum location (wetted surface and base-metal inside surface)
- Circumferential weld "AB"
- Longitudinal welds: "BA", "BB", "BC" and "BD"
- 1/4T, 1/2T, 3/4T and outside surface (T= thickness of vessel wall)
- Average neutron flux for remaining surveillance capsules

The full power flux at each of these locations was calculated based on a fuel cycle representative of an equilibrium cycle for the post-uprate time period.. Then, the corresponding fluence at each location was calculated by computing the product of each flux by the appropriate effective full power time.

7.5.3 Acceptance Criteria

A calculated neutron fluence is acceptable if its subsequent use in calculations of reactor vessel integrity yields acceptable changes in material properties (See Section 5.1.2 of the NSSS Licensing Report (LR)).

Neutron fluence at the inner surface of the reactor vessel wall (the " inner wetted surface") was calculated using the DORT computer code as described in Draft Regulatory Guide 1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (6/96).

Standard Review Plan 5.3.3, "Reactor Vessel Integrity", endorses the method contained in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," for calculating fluence at any depth of the reactor vessel wall. The method in the Regulatory Guide was used to determine the fluences at various wall depths that were used in the vessel integrity calculations (NSSS LR Section 5.1.2).

7.5.4 Results

The following table provides the neutron fluences (n/cm^2) calculated for the SGR/Uprate conditions. These values are based on 36 effective full power years (EFPY) of operation, corresponding to a 90% capacity factor over the 40 year plant operating license.

Reactor Vessel Wall Depth	**DORT Computer Program**	**DORT Computer Program with Reg. Guide 1.99 Attenuation**
Inner Wetted Surface	4.651E+19	4.651E+19
Clad-Base Metal Interface	4.590E+19	*
¼ T	2.663E+19	2.835E+19
½ T	1.357E+19	*
¾ T	6.614E+18	1.119E+19
Outside Surface	2.889E+18	*

* Adjusted Nil Ductility Transition Reference Temperature (ART_{NDT}), and therefore fluence, was not calculated for this wall depth.

Details of the derivation of the above information are provided in Reference 3, which is attached with this Licensing Report Section.

The results of the reactor vessel integrity analysis (NSSS LR Section 5.1.2) indicate that, for the analyzed fuel load configuration, the SGR/Uprate will not cause any significant increase in the EOL vessel fluence at any of the locations of interest (including welds). There are several parameters that affect the flux distribution that can be cycle-specific. However, this analysis considered a fuel cycle representative of an equilibrium cycle for the post-uprate time period.

7.5.6 Conclusions

Based on the satisfactory results of the reactor vessel integrity evaluations (NSSS LR Section 5.1.2), the SGR/Uprate will not cause any significant increase in the EOL vessel fluence at any of the locations of interest (including welds). The HNP reactor vessel is acceptable for plant operation at SGR/Uprate conditions.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

7.5.7 References

1. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials,"(Rev. 2, 5/88)

2. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"

3. BAW-2355, "Supplement to the Analysis of Capsule X Carolina Power and Light Company, Shearon Harris Nuclear Power Plant," Framatome Technologies, Inc., (Supplement 1, 11/99)

4. BAW-2241P, "Fluence and Uncertainty Methodologies," Framatome Technologies, Inc., (Revision 1, 4/99)

5. Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (6/96)

6. Standard Review Plan 5.3.3, "Reactor Vessel Integrity", Rev. 1, July 1981

BAW-2355, Supplement 1
November 1999

Supplement to the Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant

-- Reactor Vessel Material Surveillance Program --

by

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FTI Document No. 77-2355-01
(See Section 6 for document signatures.)

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FRAMATOME TECHNOLOGIES

Executive Summary

This document supplements the FTI Document BAW-2355 and provides the evaluation of the implementation of a 4.5% (to 2900 MWt) power uprate for the Carolina Power and Light Company's (CP&L) Shearon Harris Nuclear Power Plant (HNP) reactor vessel commencing with Cycle 11. The purpose of this evaluation is to document the impact of the power uprate on the HNP reactor vessel surveillance program (RVSP). The fast neutron flux and neutron fluence projections have been calculated for the HNP reactor vessel beltline materials resulting from the implementation of the power uprate in the beginning of Cycle 11. Based on the power uprate calculated flux projections and a 90% capacity factor, the projected end-of-license (36 EFPY) peak fast fluence at the clad-base metal interface of the HNP reactor vessel is 4.590 x 10^{19} n/cm^2. Using the power uprate calculated fluence and flux projections, the withdrawal schedule for the remaining HNP surveillance capsules and the HNP reactor vessel fracture toughness properties have been evaluated.

In accordance with Code of Federal Regulations, Title 10, Part 50.61, (10 CFR 50.61), the HNP reactor vessel beltline materials will not exceed the pressurized thermal shock (PTS) screening criteria before end-of-license (36 EFPY) using the power uprate fluence projections. In addition, the upper-shelf energies of the HNP reactor vessel beltline materials are not predicted to fall below 50 ft-lb at end-of-license (36 EFPY) in accordance with Regulatory Guide 1.99, Revision 2, using the power uprate fluence projections.

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1. Introduction

BAW-2355[1] presents the results of the examination of the third capsule (Capsule X) of the Carolina Power and Light Company's (CP&L) Shearon Harris Nuclear Power Plant (HNP) as part of their reactor vessel surveillance program (RVSP). This supplementary document to BAW-2355 presents the evaluation of the implementation of a 4.5% (to 2900 MWt) power uprate for the HNP reactor vessel commencing with cycle 11.

The fast neutron flux and neutron fluence projections have been calculated for the HNP reactor vessel beltline materials resulting from the implementation of the power uprate in the beginning-of-cycle 11. Based on the HNP power uprate fluence projections, the withdrawal schedule for the remaining HNP surveillance capsules have been evaluated using approved procedures and established methods and techniques in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50, (10 CFR 50) Appendix H.[2] In addition, the HNP reactor vessel fracture toughness properties have been evaluated using established methods and techniques in accordance with the requirements of Regulatory Guide 1.99, Revision 2[3] and the Code of Federal Regulations, Title 10, Part 50.61 (10 CFR 50.61).[4]

2. Neutron Fluence

2.1. Background Discussion and Objectives

CP&L is implementing a power uprate of 4.5% (to 2900MWt) beginning in cycle 11 for HNP.

To increase power, it is necessary to place once-burned assemblies on or close to the core periphery. This generally results in an increase in the relative power in the peripheral assemblies, and therefore an increase in the fast leakage flux. The power uprate configuration will also result in changes to the thermal-hydraulic parameters of the primary coolant system, which causes an increase in water temperatures, and put fresh (unburned assemblies – one side) close to, or on the core periphery. Higher water temperatures result in a lower neutron thermalization rate, and thus an increase in the fast (E > 1.0 MeV) leakage flux. These increases in fast leakage flux result in increases in the fast fluence incident on the reactor vessel and on the surveillance capsule specimens.

The HNP will operate through 36 EFPY, and the uprate will begin with cycle 11. This means that well over half of the total 36 year fluence accumulation will occur in the post-uprate time frame.

The FTI calculational based fluence analysis methodology[5] was used to calculate the neutron fluence exposure to the pressure vessel. This methodology was developed through a full-scale benchmark experiment that was performed at the Davis-Besse Unit 1 reactor, and the methodology is described in detail in Appendix A. This analysis has been performed to obtain a realistic estimate of the increase in the end-of-license (EOL) vessel fluence that would result from implementation of the power uprate in the beginning-of-cycle 11 (BOC 11).

Explicit values of the fast flux and fluence (E > 1.0 MeV) were computed for the following locations:

- Vessel Inside Surface Maximum Location (wetted surface and base-metal inside surface)
- Circumferential Weld "AB"
- Longitudinal Welds: "BA", "BB", "BC", and "BD"
- ¼T, ½T, ¾T, and Outside Surface.

- Average Neutron Flux for Remaining Surveillance Capsules

The cycle 18 full power flux at each of these locations was calculated, and the corresponding fluence at each location was then calculated by computing the product of each flux by the appropriate effective full power time.

The calculated cycle 18 full power flux was used as the *"extrapolation flux"* since it is representative of the equilibrium cycle for power uprate conditions and is the flux used to project fluences for times beyond end-of-cycle 10 (EOC 10), reported in Tables 2-1 through 2-3.

2.2. Results

The numerical and graphical results are presented in the following Tables and Figures:

- Flux and fluence results at all points of interest in Tables 2-1 through 2-3
- Locations of flux peaks on the various welds and plates in Table 2-4
- Comparison of the cycle 1 to 10 flux to the power uprate flux
- Comparison of DORT Rθ and RZ source normalization factors
- Power uprate capsule flux and lead factor
- Pressure vessel flux profile for cycles 1 through 10 and the power uprate
- Comparison of the radial and relative radial RPDs
- Comparison of the axial and relative axial RPDs
- Azimuthal flux distributions for the IS of the barrel and the PV

In Table 2-2 and Table 2-3, the project fluences were calculated by

$$\phi t(T) = (\phi t)_{cycle1\text{-}8} + \left(\phi^{extrapolated}_{cycle9\text{-}10} \times t_{cycle9\text{-}10}\right) + \left(\phi^{extrapolated}_{cycle11\text{-}EOL} \times t_{cycle11\text{-}T}\right)$$

where

$\phi t(T)$	...	fluence at time T
$(\phi t)_{cycle1\text{-}8}$	...	cycle 1 to 8 fluence[1]
$\phi^{extrapolated}_{cycle9\text{-}10}$	...	extrapolated flux for cycles 9 and 10
$t_{cycle9\text{-}10}$	...	time for cycles 9 and 10
$\phi^{extrapolated}_{cycle11\text{-}EOL}$	...	extrapolated flux for cycles 11 to EOL
$t_{cycle11\text{-}T}$	...	time from BOC 11 to time T

The results of this analysis indicate that, for the analyzed configuration, the power uprate will not cause any significant increases in the EOL vessel fluence at any of the locations of interest (including welds). There are several parameters that affect the flux distribution that can be cycle specific, however, and this analysis only considered one configuration among many possible configurations. The design of future fuel cycles must consider the potential effects on the vessel fluence before they are implemented. This is particularly important in choosing the location of the fresh or once-burned assemblies on the periphery. If fresh or once-burned assemblies were placed on or close to the peak fluence location which deviated significantly from the peripheral power distribution analyzed for cycle 18, it is probable that the flux on the circumferential weld would increase significantly over the values determined in this analysis. The same is true for placing the assemblies close to 45 degrees, because the flux on the longitudinal welds would increase significantly. Since CP&L has stated that it intends to use fuel cycle designs similar to cycle 18 for future operations, it would not be expected that fresh assemblies would be placed at any location significantly different than those analyzed for cycle 18.

Table 2-1. Summary of Base Metal and Weld Metal Fast Flux Values at the HNP Reactor Vessel "Wetted" Inside Surface and Clad-Base Metal Interface

Weld/Shell	Type	Azimuthal Range (degrees)	Axial Location in R-Z Model (cm)[a]	Fast Neutron Flux E > 1.0 MeV ($n/cm^2/sec$)
Welds at Clad-Base Metal Interface				
AB	Circumferential	0 to 45	221.9538	3.88741E+10
BC	Longitudinal	45	221.9538 to 458.2000	1.64523E+10
BD	Longitudinal	45	221.9538 to 458.2000	1.64523E+10
BA	Longitudinal	45	0 to 221.9538	1.59737E+10
BB	Longitudinal	45	0 to 221.9538	1.59737E+10
Welds at "Wetted" Inside Surface				
AB	Circumferential	0 to 45	221.9538	3.93976E+10
BC	Longitudinal	45	221.9538 to 458.2000	1.65991E+10
BD	Longitudinal	45	221.9538 to 458.2000	1.65991E+10
BA	Longitudinal	45	0 to 221.9538	1.60611E+10
BB	Longitudinal	45	0 to 221.9538	1.60611E+10
Plate at Clad-Base Metal Interface				
Intermediate	n/a	0 to 45	458.2000 to 221.9538	4.05768E+10
Lower	n/a	0 to 45	221.9538 to 0.0000	3.93785E+10
Plate at "Wetted" Inside Surface				
Intermediate	n/a	0 to 45	458.2000 to 221.9538	4.11140E+10
Lower	n/a	0 to 45	221.9538 to 0.0000	3.98297E+10
Vessel				
"Wetted" Surface	n/a	0 to 45	0.0000 to 458.2000	4.11140E+10
Clad-Base Metal Interface	n/a	0 to 45	0.0000 to 458.2000	4.05768E+10
¼T	n/a	0 to 45	0.0000 to 458.2000	2.35490E+10
½T	n/a	0 to 45	0.0000 to 458.2000	1.20059E+10
¾T	n/a	0 to 45	0.0000 to 458.2000	5.85809E+09
Outer Surface	n/a	0 to 45	0.0000 to 458.2000	2.56700E+09

[a] The origin of the DORT RZ coordinates is 227.51 cm below the core mid-plane.

Table 2-2. Fast Neutron Fluence (E > 1.0 MeV) for the HNP Reactor Vessel "Wetted" Inside Surface

Location	Cy9-10 Flux (n/cm^2-s)	Cy11-EOL Flux (n/cm^2-s)	FLUENCE (n/cm^2) EOC 8	13 EFPY	14 EFPY	15 EFPY	16 EFPY	17 EFPY
Weld AB	3.891E+10	3.940E+10	1.160E+19	1.598E+19	1.722E+19	1.847E+19	1.971E+19	2.095E+19
Weld BC	1.479E+10	1.660E+10	4.407E+18	6.120E+18	6.644E+18	7.167E+18	7.691E+18	8.215E+18
Weld BD	1.479E+10	1.660E+10	4.407E+18	6.120E+18	6.644E+18	7.167E+18	7.691E+18	8.215E+18
Weld BA	1.441E+10	1.606E+10	4.295E+18	5.961E+18	6.468E+18	6.975E+18	7.482E+18	7.988E+18
Weld BB	1.441E+10	1.606E+10	4.295E+18	5.961E+18	6.468E+18	6.975E+18	7.482E+18	7.988E+18
Int Shell	4.058E+10	4.111E+10	1.210E+19	1.667E+19	1.796E+19	1.926E+19	2.056E+19	2.186E+19
Low Shell	3.961E+10	3.983E+10	1.181E+19	1.626E+19	1.751E+19	1.877E+19	2.003E+19	2.128E+19

Location	Cy9-10 Flux (n/cm^2-s)	Cy11-EOL Flux (n/cm^2-s)	FLUENCE (n/cm^2) 18 EFPY	19 EFPY	20 EFPY	21 EFPY	22 EFPY	23 EFPY
Weld AB	3.891E+10	3.940E+10	2.220E+19	2.344E+19	2.468E+19	2.593E+19	2.717E+19	2.841E+19
Weld BC	1.479E+10	1.660E+10	8.739E+18	9.263E+18	9.787E+18	1.031E+19	1.083E+19	1.136E+19
Weld BD	1.479E+10	1.660E+10	8.739E+18	9.263E+18	9.787E+18	1.031E+19	1.083E+19	1.136E+19
Weld BA	1.441E+10	1.606E+10	8.495E+18	9.002E+18	9.509E+18	1.002E+19	1.052E+19	1.103E+19
Weld BB	1.441E+10	1.606E+10	8.495E+18	9.002E+18	9.509E+18	1.002E+19	1.052E+19	1.103E+19
Int Shell	4.058E+10	4.111E+10	2.315E+19	2.445E+19	2.575E+19	2.704E+19	2.834E+19	2.964E+19
Low Shell	3.961E+10	3.983E+10	2.254E+19	2.380E+19	2.506E+19	2.631E+19	2.757E+19	2.883E+19

Location	Cy9-10 Flux (n/cm^2-s)	Cy11-EOL Flux (n/cm^2-s)	FLUENCE (n/cm^2) 24 EFPY	25 EFPY	36 EFPY
Weld AB	3.891E+10	3.940E+10	2.966E+19	3.090E+19	4.457E+19
Weld BC	1.479E+10	1.660E+10	1.188E+19	1.241E+19	1.817E+19
Weld BD	1.479E+10	1.660E+10	1.188E+19	1.241E+19	1.817E+19
Weld BA	1.441E+10	1.606E+10	1.154E+19	1.204E+19	1.762E+19
Weld BB	1.441E+10	1.606E+10	1.154E+19	1.204E+19	1.762E+19
Int Shell	4.058E+10	4.111E+10	3.094E+19	3.223E+19	4.651E+19
Low Shell	3.961E+10	3.983E+10	3.008E+19	3.134E+19	4.517E+19

Table 2-3. Fast Neutron Fluence (E > 1.0 MeV) for the HNP Reactor Vessel Clad-Base Metal Interface

Location	Cy9-10 Flux (n/cm^2-s)	Cy11-EOL Flux (n/cm^2-s)	FLUENCE (n/cm^2) EOC 8	13 EFPY	14 EFPY	15 EFPY	16 EFPY	17 EFPY
Weld AB	3.840E+10	3.887E+10	1.144E+19	1.577E+19	1.699E+19	1.822E+19	1.945E+19	2.067E+19
Weld BC	1.465E+10	1.645E+10	4.366E+18	6.064E+18	6.583E+18	7.102E+18	7.622E+18	8.141E+18
Weld BD	1.465E+10	1.645E+10	4.366E+18	6.064E+18	6.583E+18	7.102E+18	7.622E+18	8.141E+18
Weld BA	1.430E+10	1.597E+10	4.263E+18	5.917E+18	6.421E+18	6.925E+18	7.429E+18	7.934E+18
Weld BB	1.430E+10	1.597E+10	4.263E+18	5.917E+18	6.421E+18	6.925E+18	7.429E+18	7.934E+18
Int Shell	4.006E+10	4.058E+10	1.194E+19	1.645E+19	1.773E+19	1.901E+19	2.029E+19	2.157E+19
Low Shell	3.909E+10	3.938E+10	1.165E+19	1.605E+19	1.729E+19	1.853E+19	1.977E+19	2.102E+19
IS Max	4.006E+10	4.058E+10	1.194E+19	1.645E+19	1.773E+19	1.901E+19	2.029E+19	2.157E+19
1/4T	2.321E+10	2.355E+10	6.918E+18	9.533E+18	1.028E+19	1.102E+19	1.176E+19	1.251E+19
1/2T	1.182E+10	1.201E+10	3.522E+18	4.854E+18	5.233E+18	5.612E+18	5.991E+18	6.370E+18
3/4T	5.750E+09	5.858E+09	1.714E+18	2.362E+18	2.547E+18	2.732E+18	2.917E+18	3.102E+18
Outside Surf	2.496E+09	2.567E+09	7.439E+17	1.026E+18	1.107E+18	1.188E+18	1.269E+18	1.350E+18

Location	Cy9-10 Flux (n/cm^2-s)	Cy11-EOL Flux (n/cm^2-s)	FLUENCE (n/cm^2) 18 EFPY	19 EFPY	20 EFPY	21 EFPY	22 EFPY	23 EFPY
Weld AB	3.840E+10	3.887E+10	2.190E+19	2.313E+19	2.435E+19	2.558E+19	2.681E+19	2.803E+19
Weld BC	1.465E+10	1.645E+10	8.660E+18	9.179E+18	9.698E+18	1.022E+19	1.074E+19	1.126E+19
Weld BD	1.465E+10	1.645E+10	8.660E+18	9.179E+18	9.698E+18	1.022E+19	1.074E+19	1.126E+19
Weld BA	1.430E+10	1.597E+10	8.438E+18	8.942E+18	9.446E+18	9.950E+18	1.045E+19	1.096E+19
Weld BB	1.430E+10	1.597E+10	8.438E+18	8.942E+18	9.446E+18	9.950E+18	1.045E+19	1.096E+19
Int Shell	4.006E+10	4.058E+10	2.285E+19	2.413E+19	2.541E+19	2.669E+19	2.798E+19	2.926E+19
Low Shell	3.909E+10	3.938E+10	2.226E+19	2.350E+19	2.475E+19	2.599E+19	2.723E+19	2.847E+19
IS Max	4.006E+10	4.058E+10	2.285E+19	2.413E+19	2.541E+19	2.669E+19	2.798E+19	2.926E+19
1/4T	2.321E+10	2.355E+10	1.325E+19	1.399E+19	1.473E+19	1.548E+19	1.622E+19	1.696E+19
1/2T	1.182E+10	1.201E+10	6.749E+18	7.128E+18	7.506E+18	7.885E+18	8.264E+18	8.643E+18
3/4T	5.750E+09	5.858E+09	3.287E+18	3.472E+18	3.656E+18	3.841E+18	4.026E+18	4.211E+18
Outside Surf	2.496E+09	2.567E+09	1.431E+18	1.512E+18	1.593E+18	1.674E+18	1.755E+18	1.836E+18

Table 2-3. (cont.) Fast Neutron Fluence (E > 1.0 MeV) for the HNP Reactor Vessel Clad-Base Metal Interface

Location	Cy9-10 Flux (n/cm^2-s)	Cy11-EOL Flux (n/cm^2-s)	FLUENCE (n/cm^2) 24 EFPY	25 EFPY	36 EFPY
Weld AB	3.840E+10	3.887E+10	2.926E+19	3.049E+19	4.398E+19
Weld BC	1.465E+10	1.645E+10	1.178E+19	1.229E+19	1.801E+19
Weld BD	1.465E+10	1.645E+10	1.178E+19	1.229E+19	1.801E+19
Weld BA	1.430E+10	1.597E+10	1.146E+19	1.197E+19	1.751E+19
Weld BB	1.430E+10	1.597E+10	1.146E+19	1.197E+19	1.751E+19
Int Shell	4.006E+10	4.058E+10	3.054E+19	3.182E+19	4.590E+19
Low Shell	3.909E+10	3.938E+10	2.972E+19	3.096E+19	4.463E+19
IS Max	4.006E+10	4.058E+10	3.054E+19	3.182E+19	4.590E+19
1/4T	2.321E+10	2.355E+10	1.771E+19	1.845E+19	2.663E+19
1/2T	1.182E+10	1.201E+10	9.022E+18	9.401E+18	1.357E+19
3/4T	5.750E+09	5.858E+09	4.396E+18	4.581E+18	6.614E+18
Outside Surf	2.496E+09	2.567E+09	1.917E+18	1.998E+18	2.889E+18

Table 2-4. Peak Fast Flux Locations of Welds and Plates

Weld or Plate	Peak Location (cm) Radial (cm)	Azimuthal (degrees)	Axial[a] (cm)
AB	199.7075	0	177.3238
BC & BD	199.7075	45	188.9515
BA & BB	199.7075	45	145.4585
Intermediate Plate	199.7075	0	188.9515
Lower Plate	199.7075	0	145.4585

[a] Relative to the lower active fuel elevation

Table 2-5. Comparison of HNP Power Uprate Fluxes

Location	PVIS Ratio (Uprate / Cy 1-10)	Clad IS Ratio (Uprate / Cy 1-10)
Weld		
AB	1.012	1.012
BC	1.123	1.123
BD	1.123	1.123
BA	1.117	1.115
BB	1.117	1.115
Plate		
Intermediate	1.013	1.013
Lower	1.007	1.006
Vessel		
Inner wetted surface	1.013	-
Clad-base metal interface	1.013	-
¼T	1.015	-
½T	1.016	-
¾T	1.019	-
Outer surface	1.028	-

Table 2-6. Comparison of DORT Source Normalization Factors

DORT Model	Source Normalization Factor		Ratio (Uprate / Cycles 1-8)
	Cycles 1-10	Uprate	
Rθ	7.65504E+16	7.97087E+16	1.04126
RZ	2.23933E+20	2.33047E+20	1.04070

Table 2-7. HNP 110° Capsule Flux and Lead Factor

110° Capsule Average Neutron Flux (E > 1.0 MeV)	1.90×10^{11} n/cm^2-s
PVIS Maximum Neutron Flux (E > 1.0 MeV)	4.11×10^{10} n/cm^2-s
Lead Factor (110° Location to I. S. Maximum Location)	4.62

Figure 2-1. Fast Flux Profile for HNP Cycles 1 to 10

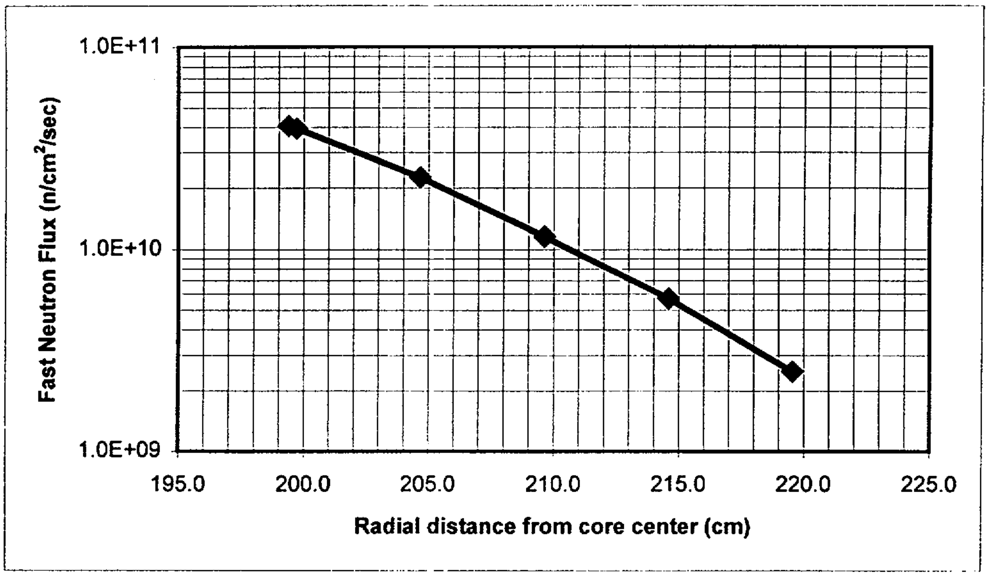


Figure 2-2. Fast Flux Profile for HNP Power Uprate

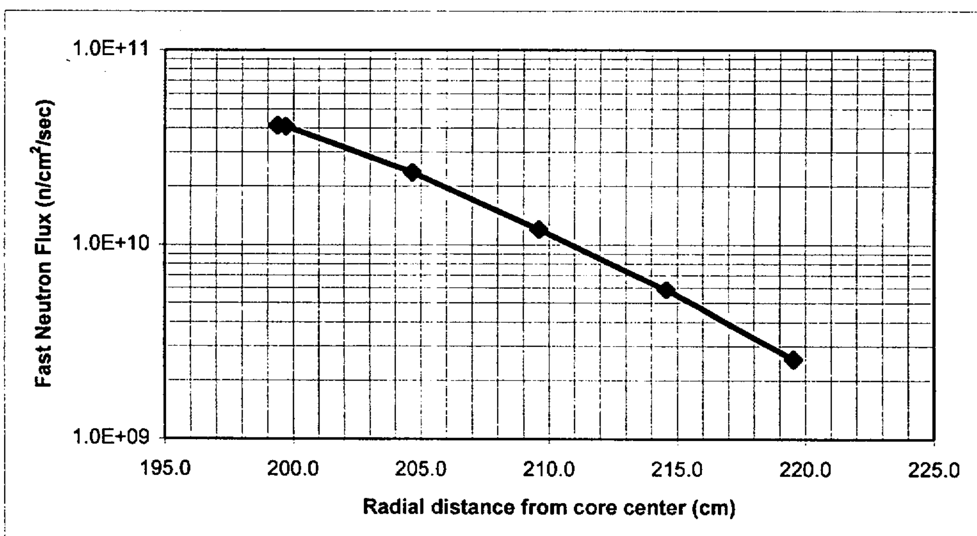


Figure 2-3. Comparison of RZ Radial Sources

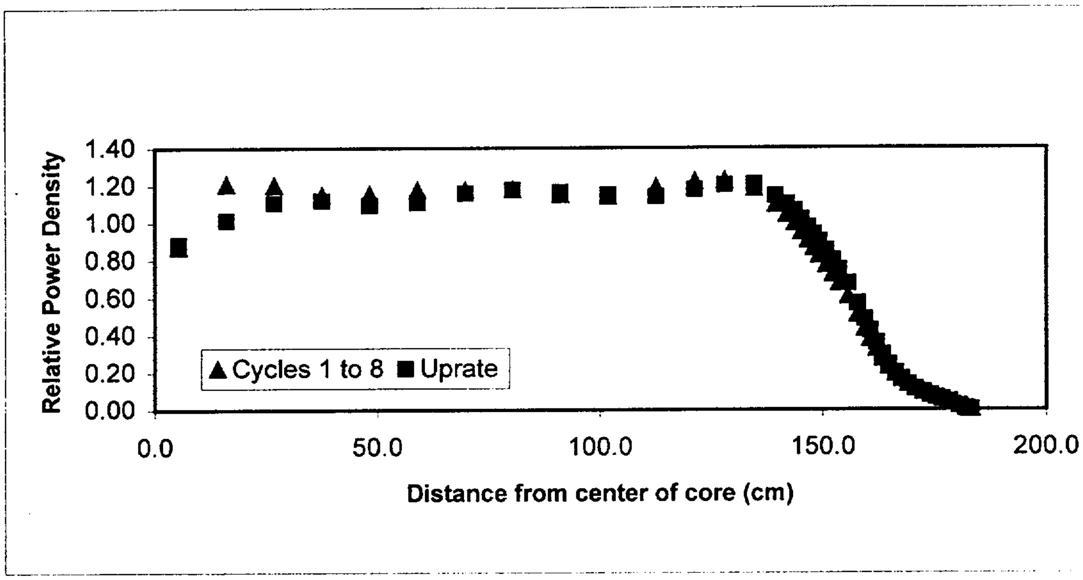


Figure 2-4. Comparison of Relative Radial RPDs

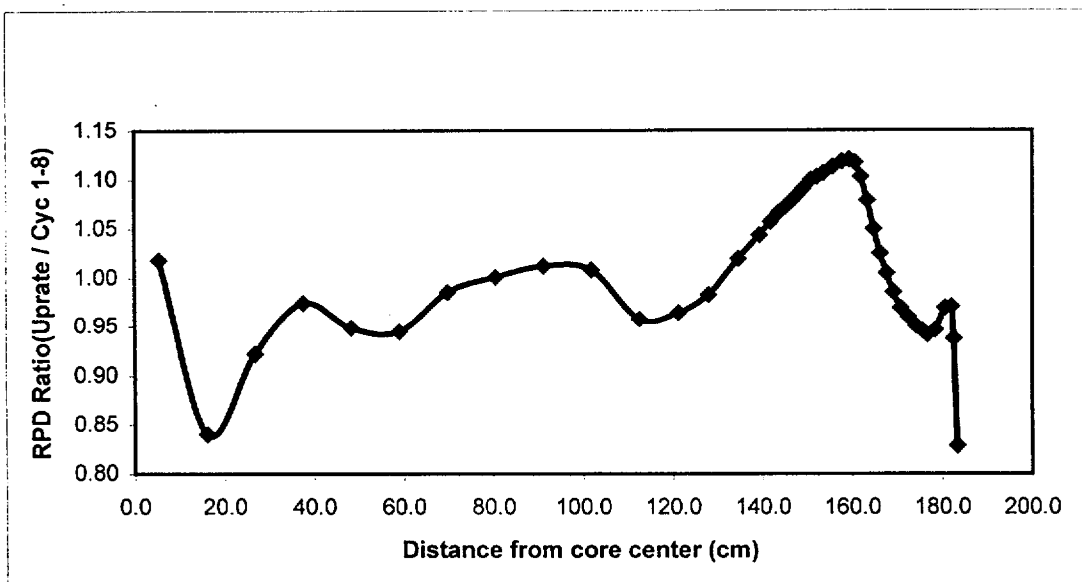


Figure 2-5. Comparison of Axial RPDs

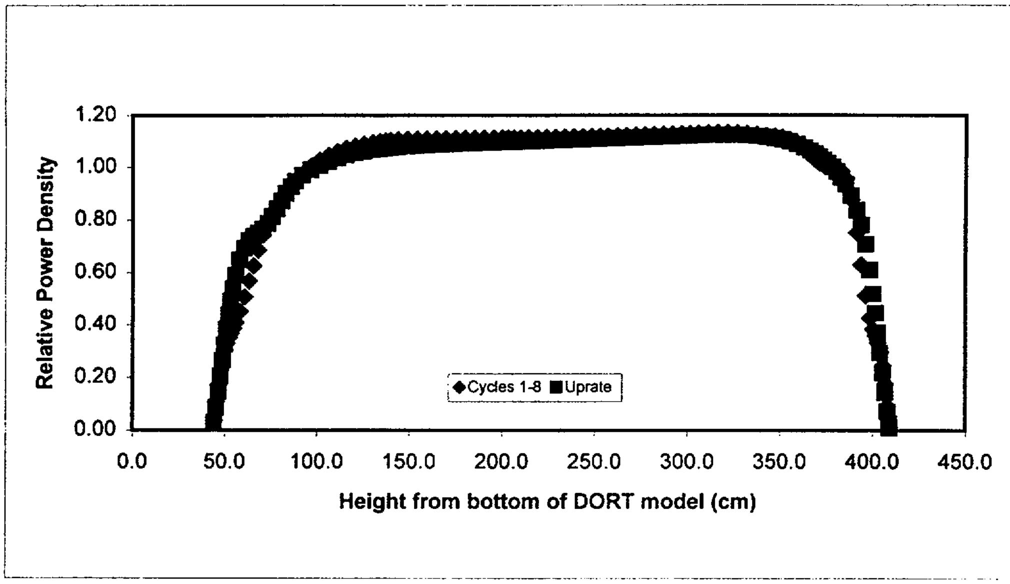


Figure 2-6. Comparison of Relative Axial RPDs

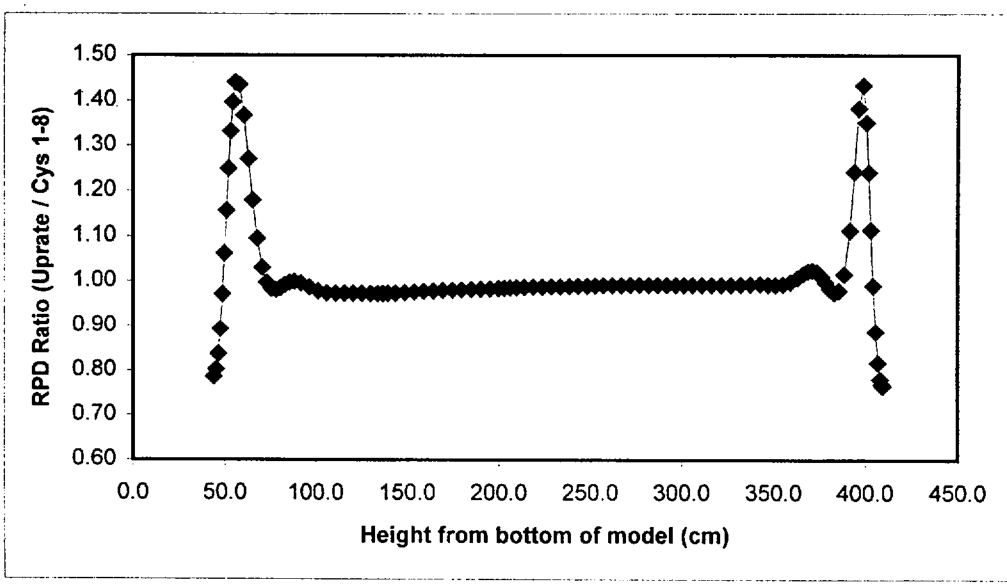
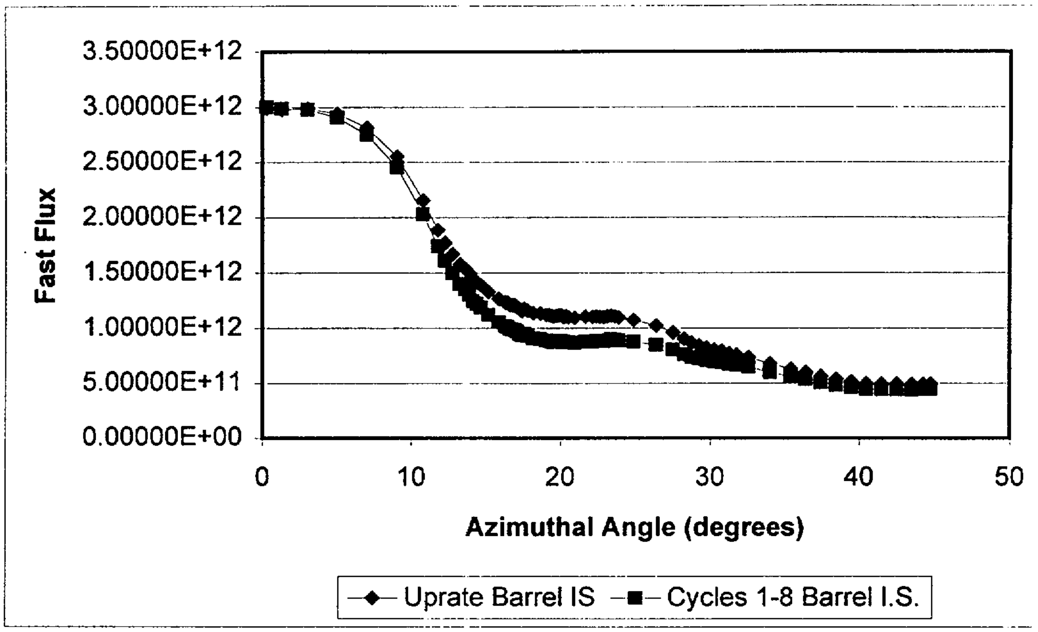
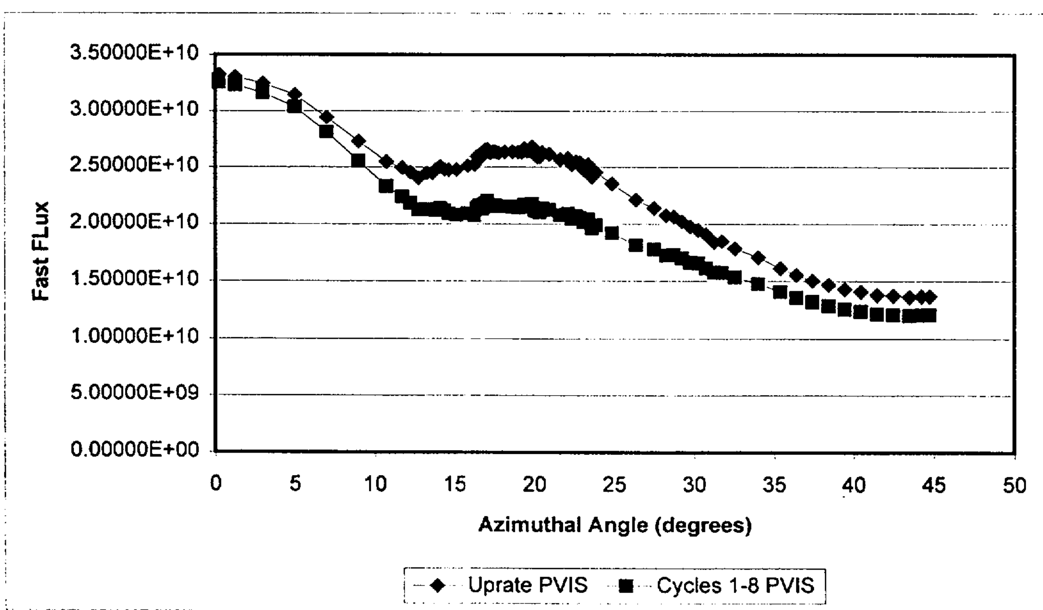


Figure 2-7. Azimuthal Flux Distribution for Core Barrel Inside Surface



Note: Flux values presented do not represent the profile at the elevation of the peak flux.

Figure 2-8. Azimuthal Flux Distribution for HNP Reactor Pressure Vessel "Wetted' Inside Surface (PVIS)



Note: Flux values presented do not represent the profile at the elevation of the peak flux.

3. Reactor Vessel Fracture Toughness

3.1. Adjusted Reference Temperature Evaluation

The adjusted reference temperatures for the HNP reactor vessel beltline region materials are calculated in accordance with Regulatory Guide 1.99, Revision 2. The adjusted reference temperatures are calculated by adding the initial RT_{NDT}, the predicted radiation-induced ΔRT_{NDT}, and the a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to 13 through 25 and 36 effective full power years (EFPY). The supporting information for the calculated neutron fluence at the "wetted" inside surface of each reactor vessel beltline material location is described in Section 2. The neutron fluence at the ¼-thickness (¼T) and ¾-thickness (¾T) wall location for each beltline material is determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., ¼T = [7.75*0.25]+0.125 = 2.0625 inches and ¾T = [7.75*0.75]+0.125 = 5.9375 inches).[6]

The evaluations for the HNP adjusted reference temperatures were performed at the ¼T and ¾T wall location of each beltline material with chemistry factors determined from Tables 1 and 2 in Regulatory Guide 1.99, Revision 2. In addition, the chemistry factors for the intermediate shell plate, heat no. B4197-2, and the intermediate shell to lower shell circumferential weld are recalculated using the available HNP surveillance data.

The ¼T and ¾T adjusted reference temperature results for the HNP reactor vessel beltline region materials applicable to 13 through 25 and 36 EFPY are presented in Tables 3-1 through 3-14. Based on these results, the controlling beltline material for the HNP reactor vessel is the intermediate shell plate, heat no. B4197-2.

3.2. Decrease in Upper-Shelf Energy Evaluation

An evaluation of the reactor vessel end-of-license (36 EFPY) upper-shelf energy at the ¼T wall location for the HNP reactor vessel beltline materials was performed using the guidelines in Regulatory Guide 1.99, Revision 2. The supporting information for the calculated neutron fluence at the "wetted" inside surface of each reactor vessel beltline material location is described in Section 2. The neutron fluence at the ¼T wall location for each beltline material is determined by calculating the ¼T depth into the vessel and adding the minimum cladding thickness (i.e., ¼T = [7.75*0.25]+0.125 = 2.0625 inches).[6]

The evaluations for the decreases in upper-shelf energies of the HNP reactor vessel were performed at the ¼T wall location of each beltline material using the respective copper contents and Figure 2 in Regulatory Guide 1.99, Revision 2. In addition, the decreases in upper-shelf energy for the intermediate shell plate, heat no. B4197-2, and the intermediate shell to lower shell circumferential weld are recalculated using the available HNP surveillance data.

The decreases in upper-shelf energy for the HNP reactor vessel beltline materials applicable to end-of-license (36 EFPY) are presented in Table 3-15. The HNP reactor vessel beltline material with the lowest predicted upper-shelf energy is the intermediate shell plate, heat no. B4197-2, however, the predicted value for this material will not fall below the required 50 ft-lb limit.

3.3. Pressurized Thermal Shock Evaluation

A pressurized thermal shock (PTS) evaluation for the HNP reactor vessel beltline materials was performed in accordance with Code of Federal Regulation, Title 10, Part 50.61 (10 CFR 50.61). The PTS reference temperature (RT_{PTS}) values are calculated by adding the initial RT_{NDT}, the predicted radiation-induced ΔRT_{NDT}, and the a margin term to cover the uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures. The predicted radiation induced ΔRT_{NDT} is calculated using the respective reactor vessel beltline materials copper and nickel contents and the neutron fluence applicable to the HNP reactor vessel end-of-license (36 EFPY). The supporting information for the calculated neutron fluence at the clad-base metal interface on the inside surface of the reactor vessel beltline where the material in question receives the highest fluence is described in Section 2.

The evaluations for the HNP RT_{PTS} values were performed for each HNP reactor vessel beltline material with chemistry factors determined from Tables 1 and 2 in 10 CFR 50.61. In addition, the chemistry factors for the intermediate shell plate, heat no. B4197-2, and the intermediate shell to lower shell circumferential weld are recalculated using the available HNP surveillance data.

The RT_{PTS} values for the HNP reactor vessel beltline materials at end-of-license (36 EFPY) are shown in Table 3-16. The results of the PTS evaluation demonstrate that the HNP reactor vessel beltline materials will not exceed the PTS screening criteria before end-of-license (36 EFPY). The controlling beltline material for the HNP reactor vessel with respect to PTS is the intermediate shell plate, heat no. B4197-2 with a RT_{PTS} value of 196.2°F which is well below the PTS screening criterion of 270°F.

Table 3-1. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 13 EFPY

Material Description[a]				Chemical Composition[a]				13 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 13 EFPY		Margin		ART, F at 13 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	1.667E19	1.016E19	4.009E18	58.2	43.3	34.0	34.0	152.2	137.3
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	1.667E19	1.016E19	4.009E18	58.2	43.3	34.0	34.0	183.2	168.3
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	1.626E19	9.912E18	3.911E18	50.9	37.7	34.0	34.0	138.9	125.7
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	1.626E19	9.912E18	3.911E18	50.9	37.7	34.0	34.0	141.9	128.7
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	6.120E18	3.731E18	1.472E18	49.4	33.9	49.4	33.9	78.8	47.8
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	1.598E19	9.741E18	3.843E18	40.7	30.1	40.7	30.1	61.4	40.2
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	5.961E18	3.634E18	1.434E18	49.0	33.5	49.0	33.5	78.0	47.0
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	1.667E19	1.016E19	4.009E18	51.6	38.4	34.0[c]	34.0[c]	[176.6]	[163.4]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	1.598E19	9.741E18	3.843E18	48.8	36.1	28.0	28.0	56.8	44.1

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-2. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 14 EFPY

Material Description[a]				Chemical Composition[a]				14 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 14 EFPY		Margin		ART, F at 14 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	1.796E19	1.095E19	4.320E18	59.5	44.5	34.0	34.0	153.5	138.5
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	1.796E19	1.095E19	4.320E18	59.5	44.5	34.0	34.0	184.5	169.5
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	1.751E19	1.067E19	4.211E18	51.9	38.8	34.0	34.0	139.9	126.8
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	1.751E19	1.067E19	4.211E18	51.9	38.8	34.0	34.0	142.9	129.8
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	6.644E18	4.050E18	1.598E18	50.9	35.2	50.9	35.2	81.8	50.4
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	1.722E19	1.050E19	4.142E18	41.6	31.0	41.6	31.0	63.2	42.0
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	6.468E18	3.943E19	1.556E18	50.5	34.7	50.5	34.7	81.0	49.4
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	1.796E19	1.095E19	4.320E18	52.7	39.4	34.0[c]	34.0[c]	[177.7]	[164.4]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	1.722E19	1.050E19	4.142E18	49.8	37.1	28.0	28.0	57.8	45.1

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-3. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 15 EFPY

Material Description[a]				Chemical Composition[a]				15 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 15 EFPY		Margin		ART, F at 15 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	1.926E19	1.174E19	4.632E18	60.6	45.6	34.0	34.0	154.6	139.6
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	1.926E19	1.174E19	4.632E18	60.6	45.6	34.0	34.0	185.6	170.6
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	1.877E19	1.144E19	4.514E18	52.9	39.7	34.0	34.0	140.9	127.7
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	1.877E19	1.144E19	4.514E18	52.9	39.7	34.0	34.0	143.9	130.7
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	7.167E18	4.369E18	1.724E18	52.4	36.3	52.4	36.3	84.8	52.6
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	1.847E19	1.126E19	4.442E18	42.4	31.7	42.4	31.7	64.8	43.4
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	6.975E18	4.252E18	1.678E18	51.8	35.9	51.8	35.9	83.6	51.8
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	1.926E19	1.174E19	4.632E18	53.7	40.4	34.0[c]	34.0[c]	[178.7]	[165.4]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	1.847E19	1.126E19	4.442E18	50.7	38.0	28.0	28.0	58.7	46.0

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-4. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 16 EFPY

Material Description[a]				Chemical Composition[a]				16 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 16 EFPY		Margin		ART, F at 16 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.056E19	1.253E19	4.945E18	61.7	46.6	34.0	34.0	155.7	140.6
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.056E19	1.253E19	4.945E18	61.7	46.6	34.0	34.0	186.7	171.6
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.003E19	1.221E19	4.817E18	53.9	40.6	34.0	34.0	141.9	128.6
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.003E19	1.221E19	4.817E18	53.9	40.6	34.0	34.0	144.9	131.6
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	7.691E18	4.688E18	1.850E18	53.7	37.5	53.7	37.5	87.4	55.0
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	1.971E19	1.201E19	4.740E18	43.1	32.5	43.1	32.5	66.2	45.0
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	7.482E18	4.561E18	1.799E18	53.1	37.0	53.1	37.0	86.2	54.0
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.056E19	1.253E19	4.945E18	54.6	41.3	34.0[c]	34.0[c]	[179.6]	[166.3]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	1.971E19	1.201E19	4.740E18	51.6	38.9	28.0	28.0	59.6	46.9

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-5. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 17 EFPY

Material Description[a]				Chemical Composition[a]				17 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 17 EFPY		Margin		ART, F at 17 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.186E19	1.333E19	5.258E18	62.6	47.6	34.0	34.0	156.6	141.6
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.186E19	1.333E19	5.258E18	62.6	47.6	34.0	34.0	187.6	172.6
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.128E19	1.297E19	5.118E18	54.7	41.5	34.0	34.0	142.7	129.5
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.128E19	1.297E19	5.118E18	54.7	41.5	34.0	34.0	145.7	132.5
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	8.215E18	5.008E18	1.976E18	54.9	38.6	54.9	38.6	89.8	57.2
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.095E19	1.277E19	5.039E18	43.8	33.2	43.8	33.2	67.6	46.4
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	7.988E18	4.869E18	1.921E18	54.3	38.1	54.3	38.1	88.6	56.2
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.186E19	1.333E19	5.258E18	55.5	42.1	34.0[c]	34.0[c]	[180.5]	[167.1]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.095E19	1.277E19	5.039E18	52.4	39.7	28.0	28.0	60.4	47.7

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-6. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 18 EFPY

Material Description[a]				Chemical Composition[a]				18 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 18 EFPY		Margin		ART, F at 18 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.315E19	1.411E19	5.568E18	63.6	48.5	34.0	34.0	157.6	142.5
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.315E19	1.411E19	5.568E18	63.6	48.5	34.0	34.0	188.6	173.5
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.254E19	1.374E19	5.421E18	55.5	42.3	34.0	34.0	143.5	130.3
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.254E19	1.374E19	5.421E18	55.5	42.3	34.0	34.0	146.5	133.3
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	8.739E18	5.327E18	2.102E18	56.0	39.5	56.0	39.5	92.0	59.0
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.220E19	1.353E19	5.339E18	44.4	33.8	44.4	33.8	68.8	47.6
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	8.495E18	5.178E18	2.043E18	55.5	39.1	55.5	39.1	91.0	58.2
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.315E19	1.411E19	5.568E18	56.3	43.0	34.0[c]	34.0[c]	[181.3]	[168.0]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.220E19	1.353E19	5.339E18	53.2	40.5	28.0	28.0	61.2	48.5

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-7. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 19 EFPY

Material Description[a]				Chemical Composition[a]				19 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 19 EFPY		Margin		ART, F at 19 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.445E19	1.490E19	5.880E18	64.4	49.4	34.0	34.0	158.4	143.4
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.445E19	1.490E19	5.880E18	64.4	49.4	34.0	34.0	189.4	174.4
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.380E19	1.451E19	5.724E18	56.3	43.0	34.0	34.0	144.3	131.0
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.380E19	1.451E19	5.724E18	56.3	43.0	34.0	34.0	147.3	134.0
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	9.263E18	5.646E18	2.228E18	57.1	40.5	56.0	40.5	93.1	61.0
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.344E19	1.429E19	5.638E18	45.1	34.4	45.1	34.4	70.2	48.8
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	9.002E18	5.487E18	2.165E18	56.6	40.1	56.0	40.1	92.6	60.2
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.445E19	1.490E19	5.880E18	57.1	43.7	34.0[c]	34.0[c]	[182.1]	[168.7]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.344E19	1.429E19	5.638E18	54.0	41.2	28.0	28.0	62.0	49.2

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-8. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 20 EFPY

Material Description[a]				Chemical Composition[a]				20 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 20 EFPY		Margin		ART, F at 20 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.575E19	1.570E19	6.193E18	65.3	50.2	34.0	34.0	159.3	144.2
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.575E19	1.570E19	6.193E18	65.3	50.2	34.0	34.0	190.3	175.2
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.506E19	1.528E19	6.027E18	57.0	43.8	34.0	34.0	145.0	131.8
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.506E19	1.528E19	6.027E18	57.0	43.8	34.0	34.0	148.0	134.8
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	9.787E18	5.966E18	2.354E18	58.1	41.4	56.0	41.4	94.1	62.8
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.468E19	1.504E19	5.936E18	45.6	35.0	45.6	35.0	71.2	50.0
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	9.509E18	5.796E18	2.287E18	57.6	40.9	56.0	40.9	93.6	61.8
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.575E19	1.570E19	6.193E18	57.8	44.5	34.0[c]	34.0[c]	[182.8]	[169.5]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.468E19	1.504E19	5.936E18	54.6	41.9	28.0	28.0	62.6	49.9

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-9. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 21 EFPY

Material Description[a]				Chemical Composition[a]				21 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 21 EFPY		Margin		ART, F at 21 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.704E19	1.648E19	6.503E18	66.0	51.0	34.0	34.0	160.0	145.0
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.704E19	1.648E19	6.503E18	66.0	51.0	34.0	34.0	191.0	176.0
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.631E19	1.604E19	6.328E18	57.6	44.5	34.0	34.0	145.6	132.5
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.631E19	1.604E19	6.328E18	57.6	44.5	34.0	34.0	148.6	135.5
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.031E19	6.285E18	2.480E18	59.2	42.3	56.0	42.3	95.2	64.6
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.593E19	1.581E19	6.236E18	46.2	35.6	46.2	35.6	72.4	51.2
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.002E19	6.108E18	2.410E18	58.6	41.8	56.0	41.8	94.6	63.6
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.704E19	1.648E19	6.503E18	58.5	45.2	34.0[c]	34.0[c]	[183.5]	[170.2]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.593E19	1.581E19	6.236E18	55.3	42.6	28.0	28.0	63.3	50.6

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-10. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 22 EFPY

Material Description[a]				Chemical Composition[a]				22 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 22 EFPY		Margin		ART, F at 22 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.834E19	1.728E19	6.816E18	66.7	51.8	34.0	34.0	160.7	145.8
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.834E19	1.728E19	6.816E18	66.7	51.8	34.0	34.0	191.7	176.8
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.757E19	1.681E19	6.631E18	58.3	45.1	34.0	34.0	146.3	133.1
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.757E19	1.681E19	6.631E18	58.3	45.1	34.0	34.0	149.3	136.1
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.083E19	6.602E18	2.605E18	60.1	43.1	56.0	43.1	96.1	66.2
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.717E19	1.656E19	6.535E18	46.7	36.1	46.7	36.1	73.4	52.2
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.052E19	6.413E18	2.530E18	59.5	42.6	56.0	42.6	95.5	65.2
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.834E19	1.728E19	6.816E18	59.1	45.9	34.0[c]	34.0[c]	[184.1]	[170.9]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.717E19	1.656E19	6.535E18	55.9	43.3	28.0	28.0	63.9	51.3

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-11. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 23 EFPY

Material Description[a]				Chemical Composition[a]				23 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 23 EFPY		Margin		ART, F at 23 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	2.964E19	1.807E19	7.129E18	67.4	52.5	34.0	34.0	161.4	146.5
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	2.964E19	1.807E19	7.129E18	67.4	52.5	34.0	34.0	192.4	177.5
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	2.883E19	1.757E19	6.934E18	58.9	45.7	34.0	34.0	146.9	133.7
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	2.883E19	1.757E19	6.934E18	58.9	45.7	34.0	34.0	149.9	136.7
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.136E19	6.925E18	2.732E18	61.0	43.9	56.0	43.9	97.0	67.8
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.841E19	1.732E19	6.833E18	47.2	36.6	47.2	36.6	74.4	53.2
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.103E19	6.724E18	2.653E18	60.5	43.5	56.0	43.5	96.5	67.0
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	2.964E19	1.807E19	7.129E18	59.7	46.5	34.0[c]	34.0[c]	[184.7]	[171.5]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.841E19	1.732E19	6.833E18	56.5	43.8	28.0	28.0	64.5	51.8

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-12. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 24 EFPY

Material Description[a]				Chemical Composition[a]				24 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 24 EFPY		Margin		ART, F at 24 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	3.094E19	1.886E19	7.441E18	68.1	53.2	34.0	34.0	162.1	147.2
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	3.094E19	1.886E19	7.441E18	68.1	53.2	34.0	34.0	193.1	178.2
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	3.008E19	1.834E19	7.234E18	59.5	46.4	34.0	34.0	147.5	134.4
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	3.008E19	1.834E19	7.234E18	59.5	46.4	34.0	34.0	150.5	137.4
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.188E19	7.242E18	2.857E18	61.8	44.7	56.0	44.7	97.8	69.4
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	2.966E19	1.808E19	7.133E18	47.7	37.1	47.7	37.1	75.4	54.2
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.154E19	7.034E18	2.775E18	61.3	44.2	56.0	44.2	97.3	68.4
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	3.094E19	1.886E19	7.441E18	60.3	47.1	34.0[c]	34.0[c]	[185.3]	[172.1]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	2.966E19	1.808E19	7.133E18	57.1	44.4	28.0	28.0	65.1	52.4

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-13. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 25 EFPY

Material Description(a)				Chemical Composition(a)				25 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 25 EFPY		Margin		ART, F at 25 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}(a)	Chemistry Factor	Inside Surface	T/4 Location(b)	3/4T Location(b)	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	3.223E19	1.965E19	7.752E18	68.7	53.9	34.0	34.0	162.7	147.9
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	3.223E19	1.965E19	7.752E18	68.7	53.9	34.0	34.0	193.7	178.9
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	3.134E19	1.910E19	7.538E18	60.0	47.0	34.0	34.0	148.0	135.0
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	3.134E19	1.910E19	7.538E18	60.0	47.0	34.0	34.0	151.0	138.0
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.241E19	7.565E18	2.985E18	62.7	45.5	56.0	45.5	98.7	71.0
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	3.090E19	1.884E19	7.432E18	48.1	37.6	48.1	37.6	76.2	55.2
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.204E19	7.339E18	2.896E18	62.1	44.9	56.0	44.9	98.1	69.8
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	3.223E19	1.965E19	7.752E18	60.9	47.8	34.0(c)	34.0(c)	[185.9]	[172.8]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	3.090E19	1.884E19	7.432E18	57.6	45.0	28.0	28.0	65.6	53.0

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-14. Adjusted Reference Temperature Evaluation for the HNP Reactor Vessel Beltline Materials at the ¼-Thickness and ¾-Thickness Locations Applicable Through 36 EFPY

Material Description[a]				Chemical Composition[a]				36 EFPY Fluence, n/cm²			ΔRT_{NDT}, F at 36 EFPY		Margin		ART, F at 36 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Intermediate Shell Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	+60	58.0	4.651E19	2.835E19	1.119E19	74.1	59.8	34.0	34.0	168.1	153.8
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	58.0	4.651E19	2.835E19	1.119E19	74.1	59.8	34.0	34.0	199.1	184.8
Lower Shell Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	+54	51.0	4.517E19	2.753E19	1.086E19	64.8	52.2	34.0	34.0	152.8	140.2
Lower Shell Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	+57	51.0	4.517E19	2.753E19	1.086E19	64.8	52.2	34.0	34.0	155.8	143.2
IS Longit. Welds (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.817E19	1.108E19	4.370E18	70.0	52.4	56.0	52.4	106.0	84.8
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	41.0	4.457E19	2.717E19	1.072E19	51.9	41.8	51.9	41.8	83.8	63.6
LS Longit. Welds (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	0.91	-20	68.0	1.762E19	1.074E19	4.238E18	69.4	51.8	56.0	51.8	105.4	83.6
Regulatory Guide 1.99, Revision 2, Position 2.1																
Intermediate Shell Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	+91	51.4	4.651E19	2.835E19	1.119E19	65.6	53.0	34.0[c]	34.0[c]	[190.6]	[178.0]
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	0.94	-20	49.1	4.457E19	2.717E19	1.072E19	62.2	50.0	28.0	28.0	70.2	58.0

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in. and "x" for ¾T = 5.9375 in.).[6]

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the ¼T and ¾T ART values

[ ] - Controlling values of the adjusted reference temperatures.

Table 3-15. Evaluation of Upper-Shelf Energy Decreases for the HNP Reactor Vessel Beltline Materials Applicable Through 36 EFPY

Material Description[a]				Cu wt%	¼T Fluence[b] (x 10^{19}) n/cm^2	Initial C_VUSE[a] ft-lbs	Predicted C_VUSE Per R.G. 1.99/2	
Reactor Vessel Beltline Region Location	Material Identification	Heat Number	Type				C_VUSE ft-lbs	% Decrease
Intermediate Shell Plate (IS)	A9153-1	A9153-1	SA-533 Gr. B1	0.09	2.84	83	63.9	23.0
Intermediate Shell Plate (IS)	B4197-2	B4197-2	SA-533 Gr. B1	0.09	2.84	71	54.4[c]	23.3[c]
Lower Shell Plate (LS)	C9924-1	C9924-1	SA-533 Gr. B1	0.08	2.75	98	76.8	21.6
Lower Shell Plate (LS)	C9924-2	C9924-2	SA-533 Gr. B1	0.08	2.75	88	69.0	21.6
IS Longit. Weld (Both 100%)	BC/BD	4P4784	ASA/Linde 124	0.05	1.11	94	75.7	19.5
IS to LS Circ. Weld (100%)	AB	5P6771	ASA/Linde 124	0.03	2.72	80	59.1[c]	26.1[c]
LS Longit. Weld (Both 100%)	BA/BB	4P4784	ASA/Linde 124	0.05	1.07	94	75.8	19.3

(a) See BAW-2355.[1]

(b) Calculated based on guidelines in Regulatory Guide 1.99, Revision 2.[3] The inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel (Table 2-2). The ¼T location fluence value is determined by calculating the ¼T depth into the vessel and adding the minimum cladding thickness (i.e., "x" for ¼T = 2.0625 in.).[6]

(c) Calculated using surveillance data in accordance with Regulatory Guide 1.99, Revision 2, Position 2.2 (i.e., fitting the surveillance data with a line drawn parallel to the existing lines in Figure 2 as the upper bound of all the data).

Table 3-16. Evaluation of Pressurized Thermal Shock Reference Temperatures for the HNP Reactor Vessel Beltline Materials Applicable Through 36 EFPY

Material Description[a]				Chemical Composition[a]									
Reactor Vessel Beltline Region Matl.	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Chem. Factor	Initial RT_{NDT},[a] F	36 EFPY Fluence at Clad-Base Metal Interface,[b] n/cm^2	Fluence Factor	ΔRT_{PTS}, F	Margin, F	RT_{PTS}, F	Screening Criteria
RT_{PTS} Calculation Per 10 CFR 50.61 Using Tables													
Intermediate Shell (IS) Plate	A9153-1	A9153-1	SA-533 Gr. B1	0.09	0.46	58.0	60	4.59E+19	1.385	80.3	34.0	174.3	270
Intermediate Shell (IS) Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	58.0	91	4.59E+19	1.385	80.3	34.0	205.3	270
Lower Shell (LS) Plate	C9924-1	C9924-1	SA-533 Gr. B1	0.08	0.47	51.0	54	4.46E+19	1.379	70.3	34.0	158.3	270
Lower Shell (LS) Plate	C9924-2	C9924-2	SA-533 Gr. B1	0.08	0.47	51.0	57	4.46E+19	1.379	70.3	34.0	161.3	270
IS Longit. Welds (Both 100%)	BC/BD	4P4784	Linde 124	0.05	0.91	68.0	-20	1.80E+19	1.161	78.9	56.0	114.9	270
IS to LS Circ. Weld (100%)	AB	5P6771	Linde 124	0.03	0.94	41.0	-20	4.40E+19	1.376	56.4	56.0	92.4	300
LS Longit. Welds (Both 100%)	BA/BB	4P4784	Linde 124	0.05	0.91	68.0	-20	1.75E+19	1.154	78.5	56.0	114.5	270
RT_{PTS} Calculation Per 10 CFR 50.61 Using Surveillance Data													
Intermediate Shell (IS) Plate	B4197-2	B4197-2	SA-533 Gr. B1	0.09	0.50	51.4	91	4.59E+19	1.385	71.2	34.0[c]	[196.2]	270
IS to LS Circ. Weld (100%)	AB	5P6771	Linde 124	0.03	0.94	49.1	-20	4.40E+19	1.376	67.6	28.0	75.6	300

(a) See BAW-2355.[1]

(b) The inside surface fluence is the calculated value at the clad – base metal interface of the reactor vessel; attenuation through the cladding is based on deterministic methods (Table 2-3).

(c) Since two of the six surveillance data points are not credible, a full margin value is used to calculate the RT_{PTS} value.

[ ] – Limiting reactor vessel beltline material in accordance with 10 CFR 50.61.[4]

4. Summary of Results

The analysis for the implementation of a power uprate of 4.5% (to 2900 MWt) for the HNP beginning in cycle 11 led to the following conclusions:

1. The projected end-of-license (36 EFPY) peak fast fluence at the clad-base metal interface of the HNP reactor vessel is 4.590 x 10^{19} n/cm^2 (E > 1.0 MeV). The corresponding fluences at the ¼T, ½T, ¾T, and outside surface vessel wall in this peak location are 2.663 x 10^{19}, 1.357 x 10^{19}, 6.614 x 10^{18}, and 2.889 x 10^{18} n/cm^2 (E > 1.0 MeV) respectively.

2. Based on the ¼T and ¾T adjusted reference temperature results calculated in accordance with Regulatory Guide 1.99, Revision 2, the controlling beltline material for the HNP reactor vessel applicable to 13 through 25 and 36 EFPY is the intermediate shell plate, heat no. B4197-2.

3. In accordance with Regulatory Guide 1.99, Revision 2, the C_VUSE values for the HNP reactor vessel beltline materials are not predicted to fall below 50 ft-lb at end-of-license (36 EFPY).

4. In accordance with 10 CFR 50.61, the HNP reactor vessel beltline materials will not exceed the PTS screening criteria before end-of-license (36 EFPY).

5. Surveillance Capsule Removal Schedule

Based on the uprated power level evaluation for the HNP reactor vessel, the following schedule is recommended for the examination of the remaining capsules in the HNP reactor vessel surveillance program:

Withdrawal / Evaluation Schedule Based on Uprated Power Level Evaluation[a]				
Capsule Identification	Location of Capsules	Lead Factor	Removal Time	Expected Capsule Fluence (n/cm^2)[b]
W	110°	2.38[c] 4.62[d]	18 EFPY[e]	6.895×10^{19} [f]
Y	290°	2.38[c] 4.62[d]	Standby	-----
Z	340°	2.38[c] 4.62[d]	Standby	-----

(a) In accordance with ASTM Standard E 185-82.[6]

(b) Based on 4.5% uprated power level evaluation (Section 2).

(c) Factor by which the capsule fluence leads the vessel's maximum inner wall fluence for cycles 1 through 10.

(d) Factor by which the capsule fluence leads the vessel's maximum inner wall fluence for cycles 11 through EOL.

(e) Approximate fluence not less than peak EOL vessel fluence at clad-base metal interface (4.590×10^{19} n/cm^2) or greater than twice the peak EOL vessel fluence at clad-base metal interface (9.180×10^{19} n/cm^2). Therefore, actual capsule removal times can range from 13.66 EFPY to 21.32 EFPY. This capsule may be held without testing following withdrawal.

(f) The specified fluence represents the peak inside surface vessel fluence at the clad-base metal interface after 60 calendar year (54 EFPY) of operation based on the current fluence estimates for plant license renewal consideration.

6. Certification

The analysis for the implementation of a power uprate of 4.5% (to 2900 MWt) for the HNP reactor vessel beginning in cycle 11 was evaluated using accepted techniques and established standard methods and procedures in accordance with the requirements of 10 CFR 50, Appendix H.

11/5/99	
M. J. DeVan (Material Analysis) Materials & Structural Analysis Unit	Date
6 Nov 99	
S. Q. King (Fluence Analysis) Performance Analysis Unit	Date

This report has been reviewed for technical content and accuracy.

11-8-99	
J. B. Hall, (Material Analysis) Materials & Structural Analysis Unit	Date
11/8/99	
E. Giavedoni (Fluence Analysis) Performance Analysis Unit	Date

Verification of independent review.

11-8-99	
K. E. Moore, Manager Materials & Structural Analysis Unit	Date

This report is approved for release.

11/8/99	
D. L. Howell Program Manager	Date

FRAMATOME TECHNOLOGIES

7. References

1. M. J. DeVan and S. Q. King, *"Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant – Reactor Vessel Material Surveillance Program –,"* BAW-2355, Framatome Technologies, Inc., Lynchburg, Virginia, October 1999.

2. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix H, Reactor Vessel Material Surveillance Program Requirements, Effective Date: January 18, 1996.

3. U.S. Nuclear Regulatory Commission, *"Radiation Embrittlement of Reactor Vessel Materials,"* Regulatory Guide 1.99, Revision 2, May 1998.

4. Code of Federal Regulations, Title 10, *"Domestic Licensing of Production and Utilization Facilities,"* Part 50.61, *"Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock,"* Effective Date: August 28, 1996.

5. J. R. Worsham III, *"Fluence and Uncertainity Methodologies,"* BAW-2241P, Revision 1, Framatome Technologies, Inc., Lynchburg, Virginia, April 1999.

6. Letter from A. R. Stalker (CP&L) to D. L. Howell (FTI), Subject: Design Input, HO-990035 (Corrected), dated May 3, 1999 (FTI Document No. 38-1247892-00).

7. ASTM Standard E 185-82, *"Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

APPENDIX A

Fluence Analysis Methodology

The primary tool used in the determination of the flux and fluence exposure to the welds, plates, and surveillance capsule specimens is the two-dimensional discrete ordinates transport code, DORT.[A-1] Carolina Power & Light Company (CP&L) has provided the cycle 18 pin x pin relative power distribution data necessary for performance of a fluence analysis in accordance with BAW-2241P, Revision 1.[A-2] Cycle 18 has been determined by CP&L to be representative of an equilibrium cycle for the post-uprate time period.

A fluence analysis was performed in accordance with BAW-2241P, Revision 1, to determine the fast flux at each location of interest.

A-1. Cycle 18 Flux Calculational Procedures

The standard Framatome Technologies, Inc. (FTI) fluence analysis procedure was used to determine the fluence accumulated in Capsule X and on the various plates and welds for cycle 18. This procedure will now be described.

Figure A-1 depicts the analytical procedure that is used to determine the incremental fluence accumulated over cycle 18. As shown in the figure, the analysis is divided into seven tasks:

(1) generation of the neutron source,
(2) development of the DORT geometry models,
(3) calculation of the macroscopic material cross sections,
(4) synthesis of the results,
(5) the calculational uncertainty, and
(6) the final fluence.

Each of these tasks is discussed below.

A-2. Generation of the Neutron Source

The time-average space- and energy-dependent neutron source for cycle 18 was calculated using the SORREL code.[A-3] The effects of burnup on the spatial distribution of the neutron source were accounted for by calculating the cycle average fission spectrum for each fissile isotope on an assembly-by-assembly basis, and by determining the cycle-average specific neutron emission rate. These data were then used with the normalized time-weighted-average pin-by-pin relative power density (RPD) distribution to determine the space- and energy-dependent neutron source. The azimuthal-average, time average axial power shape in the peripheral assemblies was used with the fission spectrum of the peripheral assemblies to determine the neutron source for the axial DORT run. These two neutron source distributions were input to DORT as indicated in Figure A-1.

A-3. DORT Analyses

The cross sections, geometry, and appropriate source were combined to create a set of DORT models (R-Θ and R-Z) for the cycle 18 analysis. Each DORT run utilized a cross section Legendre expansion of three (P_3), a minimum of forty-eight directions (S_8), and the appropriate boundary conditions. All outer boundaries employed vacuum boundary conditions. (Note that when vacuum boundary conditions are used, the location of the vacuum boundary with respect to the location of the boundary flux was checked to ensure that the boundary source is being written sufficiently far into the inner model to ensure that the boundary location does not perturb the flux significantly at the boundary flux location.) A theta-weighted flux extrapolation model was used, and all other requirements of Draft Regulatory Guide DG-1053[A-4] that relate to the various DORT parameters were met or exceeded for all DORT runs.

A-4. Synthesized Three-Dimensional Results

The DORT analyses produced two sets of two-dimensional flux distributions, one for a vertical cylinder and one for the radial plane. The vertical cylinder, which will be referred to as the R-Z plane, is defined as the plane bounded axially by the upper and lower grid plates and radially by the center of the core and a vertical line located 20 cm into the water biological shield. The horizontal plane, referred to as the R-Θ plane, is defined as the plane bounded radially by the center of the core and a point located approximately two feet into the concrete of the primary biological shield, azimuthally by the major axis, and the adjacent 45° azimuth. The vessel flux, however, varies significantly in all three cylindrical-coordinate directions (R, Θ, Z). This means that if a point of interest is outside the boundaries of both the R-Z DORT and the R-Θ DORT, the true flux cannot be determined from either DORT run. Under the assumption that the three-dimensional flux is a separable function, the two two-dimensional data sets were mathematically combined to estimate the flux at all three-dimensional points (R, Θ, Z) of interest. The synthesis procedure outlined in Draft Regulatory Guide DG-1053 forms the basis for the FTI flux-synthesis process.

A-5. Development of the Geometrical Models

The system geometry models for the mid-plane (R-Θ) DORT were developed using standard FTI interval size and configuration guidelines. The R-Θ model for the cycle 18 analysis extended radially from the center of the core to a point approximately two feet into the concrete of the primary biological shield, and azimuthally from the major axis to 45°. The surveillance capsule was modeled explicitly in the R-Θ model. The axial (R-Z) DORT geometry model was developed using FTI procedures for the radial part, and used the appropriate interval structure in the axial direction. The axial model extended from core plate to core plate. The geometrical models meet or exceed all guidance criteria concerning interval size that are provided in Draft Regulatory Guide DG-1053. In all cases, cold dimensions were used. The geometry models were input to the DORT code as indicated in Figure A-1. These models will be used in all subsequent Code of Federal Regulation, Title 10, Part 50 (10 CFR 50), Appendix H[A-5] and pressure-temperature (P-T) curve analyses.

A-6. Calculation of Macroscopic Material Cross Sections

In accordance with Draft Regulatory Guide DG-1053, the BUGLE-93[A-6] cross section library was used. The GIP code[A-7] was used to calculate the macroscopic energy-dependent cross sections for all materials used in the analysis, from the core out through the cavity and into the concrete and from core plate to core plate. The ENDF/B6 dosimeter reaction cross sections were used to generate the response functions that were used to calculate the DORT-calculated saturated specific activities.

A-7. Calculated Activities and Measured Activities

Since there was no dosimetry, the determination of C and M is not possible.

A-8. C/M Ratios

Since there was no dosimetry, the determination of C/M is not possible.

A-9. Estimation of the Best-Estimate Flux

The flux in the reactor vessel beltline region is determined by best-estimate calculations, which are, by definition, the DORT results corrected for the generic energy-dependent bias removal function. The FTI cavity dosimetry database, which was developed in the cavity dosimetry benchmark experiment, determined that there is a slight bias in the calculations. The energy-dependent bias removal function was developed to remove biases from the DORT results in order to provide best-estimate calculational results.

As discussed in the uncertainty analysis, there is no significant bias associated with this analysis beyond that identified in the Cavity Dosimetry Program. Accordingly, the energy-dependent benchmark bias function was used with the DORT-calculated flux to determine the best-estimate flux at each point of interest in the reactor vessel in accordance with the procedures discussed in the Fluence and Uncertainty Topical Report, BAW-2241P, Revision 1.

A-10. Extrapolation to the End of Life (EOL)

By necessity, extrapolation of neutron fluence to points in the future is an inexact and approximate process. It is impossible to know with certainty the character of future core operations or to accurately estimate the effect of any given core operation on the fluence at any given location, before the fact. It is possible, however, to make reasonable estimates of the inside surface maximum flux using near-future fuel cycle design trends.

The "extrapolation flux" is defined as the constant flux used to determine the fluence at points in the future. In the FTI methodology, extrapolation flux is based on the DORT-calculated flux determined in the just-completed fluence analysis. Since it is the stated intention of CP&L to continue operation after the power uprate with loadings similar to those used in the HNP cycle 18 operations, the extrapolation fluxes reported herein are appropriate and conservative.

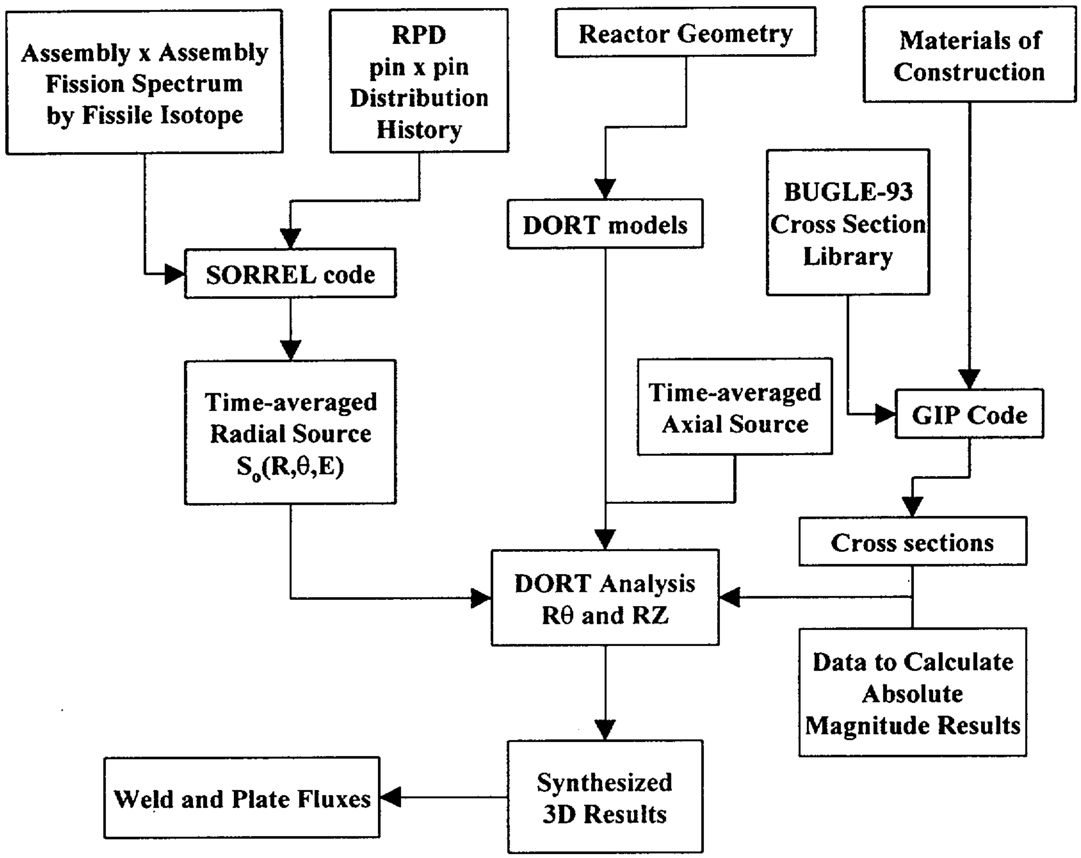
A-11. Uncertainty

The HNP reactor vessel fluence predictions are based on the methodology described in the FTI "Fluence and Uncertainty Methodologies" Topical Report.[A-2] The time-averaged fluxes, and thereby the fluences throughout the reactor, and vessel are calculated with the DORT discrete ordinates computer code using three-dimensional synthesis methods. The basic theory for synthesis is described in Section 3.0 of BAW-2241P, Revision 1, and in the previous Sections of this Appendix. The DORT three-dimensional synthesis results are the bases for the fluence predictions using the FTI "Semi-Analytical" (calculational) methodology. As noted in Sections 6.0 and 7.0 of BAW-2241P, Revision 1, the best-estimate fluence predictions are determined by removing any bias from the calculated fluence results. The bias removal function is dependent on the DORT solution procedures, the BUGLE-93 cross sections, and the FTI dosimetry benchmarks. It is independent of the HNP fluence predictions and any plant-specific comparisons of dosimetry calculations to measurements.

This analysis calculated the estimated effect of the proposed power uprate on vessel flux and fluence, but since it was performed for future operations, no dosimetry was irradiated, and thus no benchmark comparisons (C/M) are possible.

The uncertainty in the cycles 1 - 8 fluence was shown to be within the 20% NRC guideline in the base-scope analysis.[A-8] Assuming the future power distributions do not vary significantly from the cycle 18 analyzed, it can be expected that the uncertainty in the EOL fluence could be within the NRC guidelines.

Figure A-1. Fluence Analysis Methodology



A-12. References

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A-4. U.S. Nuclear Regulatory Commission, *"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,"* Draft Regulatory Guide DG-1053, June 1996.

A-5. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix H, Reactor Vessel Material Surveillance Program Requirements, Effective Date: January 18, 1996.

A-6. D. T. Ingersoll, et al., *"BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data,"* ORNL-DLC-175, Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 1994.

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A-8. M. J. DeVan and S. Q. King, *"Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant – Reactor Vessel Material Surveillance Program –,"* BAW-2355, Framatome Technologies, Inc., Lynchburg, Virginia, October 1999.

7.6 Source Terms

7.6.1 Introduction

7.6.1.1 Source Term Calculation

Source terms for accident and normal operating conditions were determined for the SGR/Uprating. The results are provided primarily for input to dose analyses and shielding evaluations. The reanalysis for the SGR/Uprating provides radiation sources for the following areas:

- Maximum Credible Accident
- Fuel Handling Accident
- Design (Reactor Coolant System [RCS]) Radiation
- Design Basis Secondary Normal Plant Operation
- N-16 Activity
- Pressurizer
- Solid Radwaste and Evaporator Concentrates
- Gas Decay Tank and Volume Control Tank
- Tritium
- Environmental Qualification
- Decay Heat

In general the power level used for calculations that are accident or "design" conditions is 2958 MWt, which is the SGR/Uprating core power plus two percent for uncertainty in determining power level. However, there are instances where the nominal core power of 2900 MWt power level is used; this is for "normal" source calculations where the basis is ANS/ANSI-18.1 and includes cases for N-16 and tritium calculations where the two percent is small compared to other variables in the analysis.

The effect of higher power is normally to increase calculated source terms. However, enrichment and cycle length are also factors, which, if changed from previous analyses, can also affect source terms on a nuclide-by-nuclide basis to either increase or decrease the individual nuclide activity.

In general, there is a direct proportionality between power level and fission product activity, which can be accounted for by scaling. Short-lived activity is proportional to power, whereas long-lived activity is proportional to power multiplied by time (burnup).

Higher enrichment results in more U-235 fission and lower thermal flux. Conversely, thermal flux is higher for lower initial enrichments at the same burnup, and thermal flux generally increases over the length of a cycle. Initial enrichment is a particularly significant parameter in explaining the thermal flux-related removal term for Xe-135. A higher thermal flux results in a higher removal rate for Xe-135, so an increasing thermal flux has the effect of decreasing the relative quantity of that nuclide.

Higher burnup results in more Pu-239 fission, higher activities of long-lived fission products and higher thermal flux. Burnup-dependent behavior includes decreasing fission yield for kryptons and bromines, attributable to plutonium build-in. In general, the effects of increasing burnup can be estimated by direct scaling of long-half-lived nuclides.

The above effects of power, burnup, and enrichment result in a complicated dependence of each nuclide on changes in these parameters; however, all are included in the computer analysis.

Changes to fuel mass affect results by means of a direct relationship to fission yields and thermal flux considerations. The effects are similar to those discussed for initial enrichment. That is, on the first order, an increase in mass is equivalent to an increase in enrichment.

The most significant "coolant activation product" during operation is N-16 (Reference 1); the activity for this nuclide will be directly proportional to the increase in power levels. This activity also varies with the time that reactor coolant takes to pass through the core and the transit time around the reactor coolant loop. The situation for activated corrosion products is similar. In this case, however, there is some degree of depletion or target burnout that may ameliorate this increase.

7.6.1.2 Computer Codes Used

Fission product inventories and decay heat were modeled with ORIGEN2,Version 2.1 (Reference 2). ORIGEN2 is a versatile point-depletion and radioactive-decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein. The ORIGEN2 code is an industry standard code based on the latest industry experimental data. In general the data are up-to-date, well documented, and accepted by the industry, and therefore, they are appropriate for the SGR/Uprating analyses.

7.6.2 Maximum Credible Accident Sources

7.6.2.1 Introduction

The HNP SGR/Uprating will provide a higher power rating for the plant and include, concurrently, the installation of new steam generators. The purpose of this report section is to present the maximum credible accident radiation sources for the uprated power level.

Calculations were performed to include variations in the power level and fuel management parameters. The selection of source terms from multiple cases is made to provide a bounding set of isotopics for use in dose calculations based on the assumptions of the Technical Information Document, TID-14844 (Reference 3).

7.6.2.2 Description of Analyses and Evaluations

Fuel management for the equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle was considered. The intent of using varied fuel management schemes for the analysis is to encompass variations that may occur in plant operation.

Calculations were made to determine the maximum credible accident radiation sources for the uprated power level. Releases are based on the release fractions described in TID-14844, 50 percent of halogens, 100 percent of noble gases, and one percent of remaining fission products, gap release fractions of Regulatory Guide 1.25 (Reference 4), release fractions of Regulatory Guide 1.4 (Reference 5) and NUREG/CR-5009 (Reference 6).

In addition, the TID-14844 released gamma energy is integrated over one year and combined with the containment free volume to provide a dose curve for confirming the Equipment Qualification (EQ) limit.

ORIGEN is a computer code system for calculating the buildup, decay, and processing of radioactive materials. ORIGEN2.1 (Reference 2) is used to calculate the fission product inventory with input describing the fuel enrichment, fuel burnup, power level, and fuel masses comprising the core. Core activities for accident source terms are taken directly from ORIGEN2.1.

Variations modeled provide a range of nuclide activities possible including an equilibrium cycle, a long equilibrium cycle, a short equilibrium cycle, axial blanket implementation, and achieving burnup at reduced power. From these calculation cases, maximum nuclide activities were selected to provide a bounding source.

7.6.2.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the Balance of Plant [BOP] Licensing Report).

7.6.2.4 Results

The results of the nuclide release calculations are used as input to the dose rate calculations. The integration of the TID-14844 release may be used to evaluate the gamma EQ radiation environment.

7.6.3 Fuel Handling Accident Sources

7.6.3.1 Introduction

This section presents the Fuel Handling Accident radiation sources for the SGR/Uprating program.

Fuel inventories are taken from equilibrium cycles with nominal, long, and short cycle lifetimes. Factors applied to inventories are based on Regulatory Guide 1.25 (Reference 4), the appropriate peaking factor for HNP, and applicable NUREG (Reference 6) data.

Using the calculated results for fission product inventory, the factors applied provide releases at 100 hours after shutdown.

7.6.3.2 Description of Analyses and Evaluations

Fuel management for the equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle was considered. The intent of using varied fuel management schemes for the analysis is to encompass variations that may occur in plant operation.

A two-fold approach was used to determine the inventory to be released in the fuel handling accident. First, a single assembly with the maximum inventory at shutdown was found for the fuel management schemes. Core activities for accident source terms were taken directly from ORIGEN2.1, which was used to calculate the fission product inventory with input describing the fuel enrichment, power level, cycle times, and fuel masses comprising the core.

In the second approach, the average assembly isotopic inventory was determined for the nominal equilibrium cycle at shutdown by dividing the whole core isotopic inventory by the number of assemblies in the core. Multiplying the average inventory by the peaking factor, it was shown that this inventory is greater than that of the maximum-inventory single assembly from the fuel management schemes and, therefore, provides a conservative basis for release calculations.

The Regulatory Guide 1.25 gap release factors were applied to the inventory of the average assembly at the end of the nominal equilibrium cycle. The gap inventory was then decayed for 100 hours. Finally, the peaking factor was applied to the inventory.

Nuclides that were not included in the inventory include those with low activity (I-129) and those with short half-lives (I-134, Kr-85m, Kr-87, Kr-88, Kr-89, Xe-137, and Xe-138) that have decayed to negligible levels at 100 hours after shutdown.

All ORIGEN2.1 runs used for inventories were made at 102 percent core power of 2900 MWt, which is consistent with thermal power uncertainties identified for HNP.

For the Fuel Handling Building, the inventory was increased to reflect 314 rods damaged in the accident. Finally, activities that include release fractions and peaking factor application for 52 BWR assemblies in the spent fuel pit that might be affected in a fuel handling accident were added to the activities calculated for the PWR fuel.

7.6.3.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.3.4 Results

The results of the nuclide release calculations are input to dose rate calculations.

7.6.4 Reactor Coolant System Radiation Sources

7.6.4.1 Introduction

This report section provides RCS radiation sources for the HNP SGR/Uprating.

For the reactor coolant system, maximum coolant activities obtained during a cycle of operation are calculated. In the calculation of maximum coolant activities obtained during a cycle of operation, small cladding defects in fuel that generate one percent of the core power are assumed to be present in each core loading and uniformly distributed throughout the core

7.6.4.2 Description of Analyses and Evaluations

Fuel management for the equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle was considered. Also considered were fuel designs that reflect the highest enrichment, as was the use of axial blankets. The intent of using varied fuel management schemes for the analysis encompasses variations that may occur in plant operation.

Parameters in the calculation of the reactor coolant fission product activities include the pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness. The minimum RCS was used and is conservative for the calculations that provide activities in microcuries per gram (μCi/g) of coolant.

It was assumed that power produced by one percent of the fuel comes from fuel with fuel defects; this is the standard basis for design sources. Also, there is no purging of the Volume Control Tank (VCT) during operation, which conservatively increases RCS activities and is consistent with Radiation Analysis Manual calculations.

The applicability of the calculation extends to 2958 MWt, which is a 2900 MWt nominal core power increased by two percent to allow for uncertainty in determining power level.

For the reactor coolant system, maximum coolant activities obtained during a cycle of operation were calculated. The ORIGEN2.1 code (Reference 2) is used to irradiate fuel through burnups attained by each fuel region in the cycle in order to determine activity inventories in the core. Inventory at intervals from zero burnup through the discharge burnup are used to determine the maximum activities that occur in the cycle.

In these calculations, small cladding defects in fuel that generates one percent of the core power are assumed to be present at initial core loading and uniformly distributed throughout the core. Similar defects are assumed to be present in all reload regions.

7.6.4.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.4.4 Results

The results of the nuclide release calculations are input to dose rate calculations or are used as input to shielding calculations.

7.6.5 Design Basis Secondary Sources

7.6.5.1 Introduction

This report section provides the design bases steam generator secondary side radiation sources for the HNP SGR/Uprating and revised RCS volume.

7.6.5.2 Description of Analyses and Evaluations

The steam generator blowdown processing system maintains the water effluent from the steam generators at a chemical and radiological specification suitable for its recycle into the main condenser or for its discharge. During normal operation, fluid from each steam generator enters under pressure into a blowdown heat exchanger, where the temperature is reduced and the blowdown is directed through the prefilter and mixed-bed demineralizers in series. After passing through outlet filters, the fluid flows through a radiation monitor and would normally be recycled to the feedwater flow, returning to the steam generator, but may be discharged to the main condenser. If the main condenser inventory reaches a high level it can be pumped to either the condensate storage tank or discharged to the environment.

The RCS radiation sources discussed in Section 7.6.4 were used as input to the calculation of steam generator secondary side activities. A value of one (a high value) is assumed for primary to secondary reactor coolant leakage. This assumption gives conservatively high values of input to the secondary side fluid.

Calculations were performed to determine the following:

- Radionuclide concentrations in the secondary side water and steam in a PWR steam generator given a primary-to-secondary leak and the operation of a steam generator blowdown processing system for cleanup
- The buildup of activity in the blowdown processing system demineralizer and filter
- The gamma ray sources for the blowdown water (secondary side water), resin and filter activities

The applicability of the calculation extends to 2958 MWt, which is a 2900 MWt nominal core power increased by two percent to allow an uncertainty in determining power level. The selection of source terms from multiple cases is made to provide a bounding set of isotopics for the secondary side given the plant and system operation parameters used.

7.6.5.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.5.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.6 Normal Plant Operation Source Terms

7.6.6.1 Introduction

The normal plant operational source terms establish the long-term concentrations of principal radionuclides in the fluid streams of the plant for subsequent use in predicting the expected release of radioactive materials from various effluent streams. The fluid streams of the plant are the reactor primary coolant and the steam generator water and steam.

Normal plant operation source terms are based on the American National Standard (ANS) Source Specification, ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Reactors" (Reference 7). The purpose of the standard, ANSI/ANS-18.1-1984, is to provide for a uniform approach, applicable to light-water-cooled nuclear power plants, for the determination of expected concentrations in fluid streams. Through application of this standard, a common basis for the determination of radioactive source terms is established with the goal of providing a consistent approach for those involved in the design, licensing, and operation of nuclear power plants.

The numerical values given in the ANSI/ANS-18.1-1984 standard are based on available data from operating plants that use Zircaloy -clad, uranium-dioxide fuel. Normal plant operational sources for HNP are established by appropriate scaling of standard values to define source term values specific to the normal plant operating parameters. The scope of the calculated values for normal plant operation sources are the values and algorithms included in the ANSI/ANS-18.1-1984.

7.6.6.2 Description of Analyses and Evaluations

The normal plant operation source term analysis uses the ANSI/ANS-18.1-1984 specifications and formulations for calculating the radionuclide activity in the fluid streams of a Light Water

Reactor (LWR) nuclear plant. The use of this standard is the accepted industry methodology for performing these calculations. The ANSI/ANS-18.1-1984 data is scaled to the proper thermal power level and specific inputs related to HNP SGR/Uprating conditions.

If the plant-specific parameters such as thermal power level, fluid system flow rates, and system fluid quantities are the same as the ANS/ANSI-18.1-1984 standard values, the source-term values of the standard are used without modification. In the case where any of the parameters differ from the values used in the standard, the source term values must be modified to account for these differences by using adjustment factors specified in ANSI/ANS-18.1-1984.

The ANSI/ANS-18.1-1984 standard is an update of the American National Standard ANSI N237 (Reference 8) based on a compilation of available operating plant data concerning primary coolant concentrations, steam generator tube leakage, and secondary side radionuclide behavior. The adjustment factors and procedures for effecting adjustments in the calculations are based on methods in ANSI N237 and retained in ANSI/ANS-18.1-1984. NUREG-0017, Revision 1 (Reference 9), uses the data values, adjustment factors, and methods in ANSI/ANS-18.1-1984. Therefore, the use of ANSI/ANS-18.1-1984 is appropriate for the update of HNP normal plant operation sources.

The calculations performed predict concentrations of the noble gases, halogens, rubidium, cesium, N-16, tritium, and other radioisotopes in the various fluid streams of HNP for normal plant operation including anticipated operational occurrences. The list of radionuclides and the concentration values predicted in this analysis are based on the current standard, ANS/ANSI-18.1-1984 (Reference 7) and are consistent with NUREG-0017, Revision 1 (Reference 9).

7.6.6.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.6.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.7 Reactor Coolant N-16 Activity

7.6.7.1 Introduction

For the SGR/Uprating program, the RCS volumes and fluid system component coolant masses for the HNP have been updated to reflect the installation of the Model Delta 75 replacement steam generators. The N-16 specific activities and loop transit times in the RCS are updated to reflect the SGR changes, as well as the uprating in power level. The predicted N-16 specific activity in the RCS is based on predicted O-16 reaction rates, primary system component volumes, and RCS temperature and fluid flow conditions for HNP.

7.6.7.2 Description of Analyses and Evaluations

The N-16 neutron activation rates or O-16 neutron reaction rates, i.e. the product of neutron flux and reaction cross-section for the $^{16}O(n,p)^{16}N$ neutron reaction, employed in the analysis are those developed for a three-loop plant with a 1.125-inch thick baffle neutron pad at a reactor thermal power of 2652 MWt. The HNP reactor internals geometry is the same as the internals geometry used in the analysis, and reaction rates are directly applicable to HNP with the only correction a direct scaling of the input data to the HNP uprated core power level of 2900 MWt.

Primary loop component volumes are used to predict primary loop transit times for the calculation of the N-16 specific activity as a function of location in the RCS loop. Volumes of the various components in the primary system are scaled to hot operating conditions for the N-16 analysis.

Calculations were performed to determine the radionuclide source terms from the buildup and decay of radioactive materials in the fluid systems. The N-16 specific activity calculation uses the primary system volumes, flow rates, flow fractions, and coolant densities at operating conditions to predict the reactor coolant N-16 specific activity at specified locations in the primary loop. Calculations of the N-16 specific activity in the pressurizer liquid and vapor volumes are also predicted in the analysis.

7.6.7.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.7.4 Results

The results of the nuclide release calculations are input to dose rate calculations or are used as input to shielding calculations.

7.6.8 Pressurizer Sources

7.6.8.1 Introduction

This report section provides the design basis pressurizer radiation sources for the SGR/Uprating and revised RCS volume.

7.6.8.2 Description of Analyses and Evaluations

The RCS sources discussed in Section 7.6.4 are used as inputs to calculations for the pressurizer sources. The RCS sources are calculated for one percent fuel defects and a core power level of 2958 MWt. The applicability of the calculation extends to 2958 MWt, which is a 2900 MWt nominal core power increased by two percent to allow an uncertainty in determining power level. The selection of input terms is made from multiple cases of RCS activity calculations to provide a bounding set of isotopics for the pressurizer activity calculation.

The pressurizer liquid specific activity is assumed to be the same as that of the reactor coolant. Pressurizer steam phase radiogas concentrations are based on the stripping of radiogases from the continuous 2-gpm pressurizer spray and the subsequent buildup of these radiogases in the steam space. The buildup time is assumed to be one year. The radiogases are assumed to be completely stripped from the spray, except for Kr-85, where a stripping fraction of 0.9 is used.

Pressurizer steam phase iodine concentrations are obtained from the liquid phase nuclide activities and measured values of the partition coefficient for I-131. A large partition coefficient was chosen to maximize the activities. It was assumed to apply to all radioiodines.

7.6.8.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.8.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.9 Solid Radwaste And Evaporator Concentrates

7.6.9.1 Introduction

This report section provides the solid waste and evaporator concentrates radiation sources for the SGR/Uprating program. Sources are generated both for normal operation and design basis.

The radiation sources are presented for the boron recycle evaporator concentrates and for demineralizers in the CVCS and boron recycle system. Other waste evaporators and demineralizers were not updated since their function has been replaced by other equipment.

7.6.9.2 Description of Analyses and Evaluations

The boron recycle evaporator bottoms activities were calculated. The activity of the CVCS mixed-bed demineralizer resins for design basis sources was calculated. The other demineralizer resin activities for both normal and design activities were also calculated. Activity on each resin was calculated using system flow rates and decontamination factors to determine the rate of radioactivity deposit on the resin. Since the cation bed is only used intermittently (between 0 and 10 percent of the time), for the mixed bed, the conservative assumption is made that all the long-lived activity is removed on the mixed bed. For the maximum activity on the cation bed, it was assumed that short-lived cations will be equal in activity to the mixed bed, and for the long-lived cations a maximum of 10 percent of the total mixed bed activity will be present.

7.6.9.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.9.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.10 Gas Decay Tank and Volume Control Tank Sources

7.6.10.1 Introduction

This report section provides the Gas Decay Tank (GDT) radiation sources and the Volume Control Tank (VCT) radiation sources at the SGR/Uprating conditions.

GDT radiation sources were determined for the shutdown of the reactor after operation for an equilibrium cycle and for a 40-year inventory of Kr-85. VCT radiation sources were determined for maximum activities of nuclides in the VCT during operation of the equilibrium cycle.

7.6.10.2 Description of Analyses and Evaluations

Gaseous Waste Processing System (GWPS) radiation sources are calculated as GDT inventory after 40 years of operation for both the design basis RCS sources and normal RCS sources. For design basis RCS activities, the RCS activities were developed with and without the operation of the GWPS (purging of the VCT during operation).

Liquid Waste Processing System (LWPS) radiation sources in FSAR Table 15.7.2-2 (and repeated in Table 15.7.2-3) are for non-seismic equipment for consideration of liquid system equipment failure and release of radioactive noble gases.

RCS and VCT activities are used as input to the GDT and VCT calculations. These are taken from an equilibrium cycle, a long equilibrium cycle, a short equilibrium cycle, high enrichment, and axial blanket implementation. There is no purging of the VCT during operation. This conservatively increases the reactor coolant and VCT activities relative to operation with purging during operation.

GDT radiation sources are calculated for shutdown of the reactor after operation for an equilibrium cycle. Maximum RCS and VCT activities for an operating cycle are taken as input.

Since no purging of the VCT has occurred in the cycle there is an inventory of noble gases in the VCT vapor. A calculation is done to simulate the purging of the VCT to the GDT at shutdown, degassing of the RCS with the maximum RCS letdown rate for 3 hours, followed by another purge to the GDT. The cycle of degassing for 3 hours and purging to the GDT is continued for a total of 10 purges. For the iodines, there is assumed a decontamination factor (DF) of 10 and a partition factor of 100 to determine the activity in the VCT vapor.

Maximum GDT (except for Kr-85) activities are selected over the degassing period for the GDT sources. The method takes this approach to maximize the activities for input to accident analysis.

The treatment of Kr-85 is different from the other isotopes. This calculation assumes that the entire RCS Kr-85 inventory produced in the cycle is situated in theGDT.

Noble gases and iodines are the nuclides of interest for the analysis of the VCT rupture. Calculations were done by using maximum RCS and VCT activities and applying factors to account for reductions in concentrations due to resin bed removal or VCT stripping.

For Kr-85, Henry's Law constant is used to determine the Kr-85 in Henry's law equilibrium with the RCS. Some small amount of noble gases is in the VCT liquid. This amount was calculated by using the RCS activity and the stripping fraction for the nuclide.

The majority of the iodines are in the VCT liquid. The specific activities, μCi/g, are calculated by applying a DF of 10 to the RCS values in order to account for removal by the mixed bed demineralizer that is upstream of the VCT.

The GWPS accumulated inventory values based on design RCS activities after one cycle are calculated by multiplying the GDT activity by the volume of the control tank, except for Kr-85. The decay of the nuclides during operation and shutdown from cycle to cycle reduces the GWPS inventory values to very small values so that the last cycle of operation provides the GWPS inventory.

Using the values of inventory, the decay of nuclides is calculated for 30 days and 50 days after shutdown. The calculation of the expected (using normal sources) accumulated radioactivity in the GWPS after forty years of operation is done by the same method, using one-cycle inventory and the 40-year buildup of Kr-85.

7.6.10.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.10.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.11 Tritium Source Calculations

7.6.11.1 Introduction

RCS coolant volumes for HNP have been updated to reflect the SGR/Uprating. Tritium generation in the RCS was updated to reflect the SGR changes, the uprating in power level, and the projected refueling plan for HNP. Analyses were performed to predict the tritium generation and distribution in the reactor coolant system based on the SGR/Uprating and projected operational plans at the HNP.

7.6.11.2 Description of Analyses and Evaluations

Tritium is produced in Light Water Reactors in several ways. The primary method is a ternary fission product from fission in the fuel rods of the active core.

Tritium from this source, along with tritium produced from boron reactions in burnable poison rods and fuel rods containing boron as a burnable absorber, must diffuse through the fuel or burnable poison material and cladding before release to the RCS. Tritium is also produced directly in the reactor coolant through nuclear reactions involving boron, lithium, and deuterium.

The utilization of burnable poison rods or fuel rods containing boron is not projected for HNP in fuel cycles beyond cycle 7, therefore, analysis of the tritium source from burnable poisons is not included in this analysis.

The analysis of the tritium sources is based on either the thermal power in the active core or the neutron flux levels in the coolant water of the active core.

The tritium source from ternary fission is based on the thermal power level of the reactor and the fission product yield for the tritium isotope. Subsequent release of the tritium to the RCS coolant water is based on either design basis or expected basis release fractions of tritium from Zircaloy-clad fuel rods.

Tritium sources from neutron reactions with elements in the coolant water are based on the projected levels of boron and lithium added to the coolant water during normal operation or the naturally occurring deuterium in the coolant water. The analyses for the three sources of tritium in the coolant water, i.e., coolant soluble boron used for reactor control, coolant soluble lithium used for pH control of the coolant water, and coolant deuterium are based on the same methodology. Neutron flux values in the coolant water regions of the active core are obtained by scaling the core-average flux values by flux ratios of region-wise fluxes from unit cell for a representative fuel pin lattice. The group-wise scaling factors correct the core-average fluxes to coolant water region fluxes. Boron, lithium, and deuterium reaction rates in the coolant water of the active core are predicted by multiplying the neutron fluxes by the reaction cross sections for the tritium producing reactions of boron, lithium, or deuterium isotopes in the coolant water.

7.6.11.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since subsequent calculations are performed for tritium releases that use these sources as inputs (see the BOP Licensing Report).

7.6.11.4 Results

Results of the tritium source analysis are used to evaluate plant tritium generation and release.

7.6.12 Decay Heat Generation

7.6.12.1 Introduction

This report section provides decay heat generation for the SGR/Uprating and revised RCS volume. Decay heat is calculated for shutdown of the reactor after long-term, steady-state operation for an equilibrium cycle at intervals useful in fluid system analysis and plant procedures.

7.6.12.2 Description of Analyses and Evaluations

The anticipated HNP fuel management strategies, which include an equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle, were considered. The intent of using varied fuel management schemes for the analysis is to encompass variations that may occur during plant operation. The analysis provides a bounding decay heat curve.

ORIGEN2.1 runs for the equilibrium cycle(s) are used to calculate a total core inventory of actinides and fission products (Reference 2). The ORIGEN2.1 calculations were performed at two-percent power above the rated core power of 2900 MWt, or 2958 MWt. Decay heat values were taken directly from ORIGEN2.1.

After the reactor is tripped, fissioning of considerable magnitude continues due to delayed neutrons for a brief time, but rapidly diminishes (after about 100 seconds) to insignificant relative to the heat produced by fission product and actinide decay. The period of interest for fluid systems analysis in calculating the Residual Heat Removal (RHR) cooldown transient is from about 4 hours after reactor shutdown to about 50 hours. Therefore, residual heat due to delayed neutron fissioning is not accounted for in this analysis since the first decay step in the ORIGEN2.1 runs documented here is one hour after shutdown.

7.6.12.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since subsequent calculations use the bounding decay heat curve as inputs (see the BOP Licensing Report).

7.6.12.4 Results

Decay heat was calculated for shutdown of the reactor after long-term, steady-state operation for an equilibrium cycle at intervals useful in fluid system analysis and plant procedures.

7.6.13 Source Term Analysis - Conclusions

Source terms for accident and normal operating conditions were determined for the SGR/Uprating. The results are provided primarily for input to dose analyses and shielding evaluations.

In general, the power level used for calculations that are accident or design conditions is 2958 MWt, which is the SGR/Uprating (core) power plus two percent for uncertainty in determining power level. However, there are instances where the nominal core power of 2900 MWt power level is used; this is for "normal" source calculations where the basis is ANSI/ANS-18.1 and includes cases for N-16 and tritium calculations where the two percent is small compared to other variables in the analysis.

Whether for design or normal radiation source calculations, the effect of higher core power is to increase calculated source terms. Therefore, the results obtained with the Model Delta 75 replacement steam generators at the uprated NSSS power of 2912.4 MWt (core power of 2900 MWt) bound operation with the Model Delta 75 replacement steam generators at the current NSSS power of 2787.4 MWt (core power of 2775 MWt).

Since radiation source terms are provided for input to dose analyses, shielding analyses or evaluations, or for plant use, the users of the source terms must determine, when appropriate, that the resulting analysis or evaluation is consistent with and continue to comply with the current HNP licensing.

7.6.14 References

1. *Nuclear Reactor Engineering*, Sections 6.189, 7.36 and 7.37, Samuel Glasstone and Alexander Sesonske, 1963.

2. CCC-371,"ORIGEN2.1: Isotope Generation and Depletion Code - Matrix Exponential Method," RSIC Computer Code Collection, Oak Ridge National Laboratory, February 1996.

3. Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," J. J. DiNunno, F. D. Anderson, R. E. Baker, R. L. Waterfield, Division of Licensing and Regulation, U. S. Atomic Energy Commission, March 23, 1962.
4. Reg. Guide 1.25 Rev. 0 "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," Revision 0, March 23, 1972.

5. Reg. Guide 1.4 Rev. 2 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss Of Coolant Accident For Pressurized Water Reactors," Revision 2, June 1974.

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

7. ANSI/ANS-18.1-1984, "American National Standard Radioactive Source Term Normal Operation of Light Water Reactors," American Nuclear Society, LaGrange Park, Illinois, Approved December 31, 1984.

8. ANS-18.1/ANSI N237-1976, "American National Standard Source Term Specification," American Nuclear Society, LaGrange Park, Illinois, Approved May 11, 1976.

9. NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1985.